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Current Status and Issues of Technology for Plutonium Use

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CONTENTS

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Japan's Perspective on Peaceful Usage of Plutonium

94FE0521A Tokyo GENSHIRYOKU KOGYO in Japanese Jan 94 pp 10-15

[Article by Y. Moriguchi, Science and Technology Agency, Nuclear Fuel Division]

[Text]

1. Introduction

It has been more than 30 years now that Japan has been producing nuclear energy. Nuclear power generation by light water reactors now accounts for 30 percent of all energy produced in Japan. The recycling of plutonium, however, which involved an overseas shipment of plutonium to Japan late last year to early this year, has aroused concern and stirred a lot of debate both inside and outside Japan. The R&D on plutonium usage has consistently been one of the most serious problems Japan has had to face since starting developing and using nuclear energy. Today, Japan finds itself poised to put its fast breeder reactor, "Monju," on line and officially begin the plutonium era.

The Atomic Energy Commission of Japan has been working on a revision of the long-term plan for developing uses of nuclear energy. The plutonium recycling plan has become one of the major topics of discussion in this revision. A special subcommittee of the commission is currently holding deliberations in order to put the final touches on the report, but the basic policies of Japan regarding plutonium use in the future will remain firm even with the round of deliberations taking place.

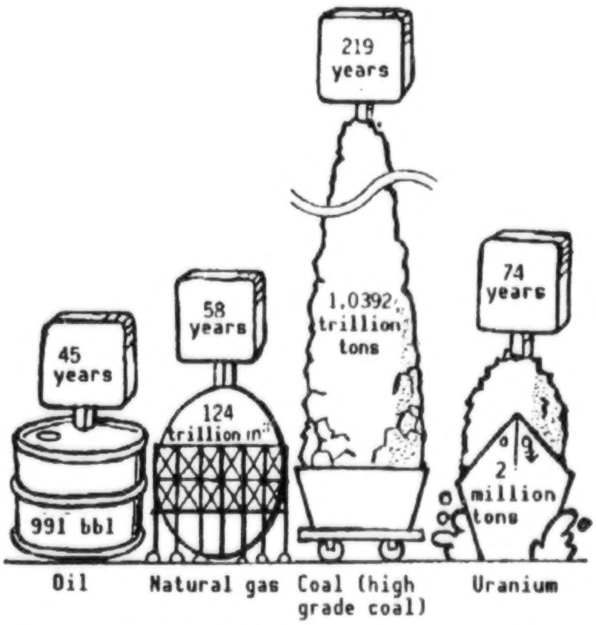
In this paper, we will in fairly simple terms describe the thinking behind the use of plutonium in Japan.

2. Necessity and Meaning of Nuclear Fuel Recycling in Japan

(1) Resource Theory

The nuclear energy produced by light water reactors accounts for approximately 30 percent of all energy generated in Japan, and when we consider its use as an energy alternative in the future, a general consensus seems to be emerging among the people of Japan that nuclear energy has an essential role to play in generating energy in the future. As soon as we accept a fixed role for nuclear energy, we are confronted with the problem of plutonium use.

In generating nuclear energy, plutonium is invariably produced. Thus, the question boils down to whether to bring technology to bear in making use of plutonium as a resource or simply disposing of it as waste. There are countries without doubt, particularly those rich in natural resources such as oil, coal, and natural gas, that have adopted policies of not using plutonium. There seems to be a strong possibility, however, that uranium deposits will last no more than 70 years or so, and even by best estimates, that resources will last no longer than when our grandchildren become adults. The same can be said for other resources such as oil and natural gas (Figure 1).



(Note 1) Reserve production ratio (Verifiable recoverable reserves/annual production)
(Note 2) Using plutonium increases the number of years uranium can be used from several to tens of times.
Source: SOGO ENERUGI TOKEI, et al.

Figure 1. Energy Resources of World

If plutonium were used as an energy resource, however, the potential amount of resources would last more than 1000 years. It would be irresponsible for this generation to be the ones who by themselves used up all the world's resources and enjoyed an advanced consumption-oriented civilization. There are also those who say there is nothing to worry about in the foreseeable future because of the low cost of uranium and the stable supply of oil. What people must keep in mind, however, is that it takes a long time to develop the technologies for plutonium use. It has already been more than 20 years, for example, since planning on the "Monju" fast breeder reactor began. If Japan, a country poor in energy resources, is to live up to its responsibility in terms of ensuring energy resources for future generations, the R&D on plutonium uses takes on whole new implications.

(2) International Responsibility

We mentioned the necessity of promoting plutonium use from the view of securing energy resources for countries poor in energy resource such as Japan, but there are many countries which have no policies regarding plutonium use. It would not be stretching matters to say, however, that those countries are for the most part either rich in energy resources or too poor economically to do R&D with a long-term perspective. Japan, on the other hand, is fortunately one of a few countries which have leading-edge technologies that enable it to perform R&D with the long-range point of view in mind. If we look with seriousness at ensuring energy resources for the people of the world in the 21st century, the major resource consuming countries such as Japan have a major role to play in contributing their technical expertise to other countries.

(3) Environmental Impact

Plutonium is invariably produced in the generation of nuclear energy. There are a number of problems in disposing of plutonium in that form. There are those who point to the toxicity and resulting danger of plutonium, but we believe that the methods of disposing spent fuel containing plutonium are environmentally safer than methods of disposing high-level radioactive vitrified waste in which plutonium has been removed. Additional research is being done on ways to remove other elements contained in high-level waste having the same super half-life as plutonium and burn these as fuel in fast breeder reactors. Doing this will continue to lessen the impact on the environment.

In terms of environmental impact, we need to find ways in which plutonium is used in a positive manner.

3. Dealing With Fears Concerning Plutonium Use

(1) Safety

There are people who say that plutonium is deadly poison, but this is because plutonium emits alpha rays when absorbed into the body and this reportedly increases the chances of cancer. Alpha rays, however, are a type of radiation which can be stopped by a single sheet of paper, and this danger can be eliminated by preventing absorption of plutonium into the body.

Japan has a good track record when it comes to safely handling plutonium, most of which is done by the Power Reactor and Nuclear Fuel Development Corporation (Table 1). If we judge by that experience, the safe handling of plutonium should be as attainable in the future as well. Japan also has a good track record when it comes to handling and making practical use of substances far more toxic than plutonium, i.e., cyanide, which it does in much larger quantities.

Table 1. Plutonium Handling Record

Operating Record of Tokai Reprocessing Plant		
Number of Assemblies Processed		
BWR	2234	(Approximate weight: 190 kg/assembly)
PWR	590	(Approximate weight: 400 kg/assembly)
ATR	170	(Approximate weight: 150 kg/assembly)
JPDR	161	(Approximate weight: 60 kg/assembly)
Manufacturing Record of Plutonium Fuel Plants		
Number of Assemblies Processed/Manufactured		
ATR	Fugen	530
FBR	Joyo	425
FBR	Monju	126

As plutonium is produced daily by the burnup of uranium in existing light water reactors, it is burned up in the reactors. In other words, nearly 30 percent of the electric power from existing nuclear power plants is obtained from plutonium. Accordingly, we could say that about 10 percent of the electricity being sent to our homes is the result of plutonium being burned. As we can see, dealing with plutonium is nothing new as we have been doing it all along, so there should be no problem in the future handling plutonium safely.

(2) Nuclear Non-Proliferation Activities

One of the reasons there is so much fear regarding the use of plutonium is that plutonium might be used as material for nuclear weapons. Domestically speaking, there are few people truly concerned about the fact that Japan is engaged in nuclear energy development. Under the current international situation, the advantages of Japan being involved in nuclear energy development far outweigh the disadvantages. Internationally, however, there are more people than we realize who are concerned about Japan developing nuclear weapons. Again, in a global context, there are several countries whose intentions the U.S. and other countries view suspiciously regarding the development of nuclear energy, thus there is a lot of talk in Japan over nuclear non-proliferation, and particularly whether Japan should abandon its policies regarding the peaceful use of plutonium. Accordingly, it is essential that Japan clearly state its commitment, both at home and abroad, regarding the peaceful use of nuclear energy.

More specifically, Japan should restate its commitment domestically to limiting research, development, and use of nuclear energy to peaceful purposes based on the Atomic Energy Act and adopt these three principles as national policies. Internationally, Japan should be a

	BNFL (U.K.)	COGEMA(France)	Purchased from U.S. and Germany	Domestic (Tokai Reprocessing Plant)
(Shipped)	2300t ¦ 1130t	2340t		724t
(Reprocessed)	0 ¦ 1100t THORP ¦ Gas furnace	150t ¦ 620t UP 2 ¦ UP 3		645t ¦ 26t ¦ 9t Power company share ¦ PNC share ¦ JAERI share
(Recovered Pu)	0kgPuf 1560kgPuf Power companies inven. 900kgPuf 320kg purchased from the U.K.	780kgPuf 2480kgPuf Power companies inven. 2000kgPuf 5kg Puf purchased from France	160kgPuf	2840kgPuf 80kgPuf 5kgPuf
	980kg Puf shipped PNC purchased 620kg Puf of that from power companies for ¥3.72 billion	1250kgPuf shipped PNC purchased 1250kg Puf of that from power companies for ¥2.07 billion • 190kg Puf for Y940 million • 1060kg Puf (Akatsuki-maru [nuclear-powered ship]) • 5 kg Puf purchased from France (does not agree with total because of rounding)		2925kgPuf recovered PNC share purchased from power companies (cumulative) • 1370kgPuf (cumulative amount as of year-end 1992)

Usage		
Monju FBR	**R&D use**	**Fugen ATR/Joyo FBR**
• PU usage 1.1t 1300kg Puf (30 Jun 93) • Initial fuel loading completion rate 54% 82% (30 Jun 93) • Initial raw plutonium loaded (verified) 1360kg Puf • Initial fueling projection 1020kg Puf • Initial MOX fuel 6.7t MOX	• 44kg (LWR (Kansai Electric Power, etc.) • 300kg (JAERI FCA, etc.) • 180kg (PNC DCA, etc.)	2050kg Puf
PNC inventory		
• Recovered by Tokai Reprocessing Plant (31 Mar 93) 570kg Puf (PNC share of that) 410kg Puf	• From overseas 1060kg Puf	

Table 1. Plutonium Supply and Demand in Japan (year-end 1992 figures unless otherwise indicated)

member of the Nuclear Non-Proliferation Treaty (NPT), and receive IAEA (International Atomic Energy Agency) safeguards regarding nuclear materials. By adopting that stance, it will become apparent to everyone that Japan has no intention of maintaining nuclear weapons, but another thing that Japan must do is ensure transparency. Doing that will show that Japan has nothing to hide with regard to plutonium use. There are restrictions on what can be revealed because doing so might increase the danger of nuclear proliferation, but basically this means releasing all information. To take one example, in October of last year the government disclosed publicly how much plutonium it was handling (Table 2). Internationally, as well, there are demands for more transparency regarding the peaceful use of plutonium. For that reason, the Science and Technology Agency has proposed an international system for managing plutonium with the primary aim being to improve transparency. We believe this could become a vehicle for exchanging ideas between related organizations inside and outside Japan (Table 3).

Table 3. Basic Framework (Summary) of International System of Control on Plutonium and Highly Enriched Uranium (Science and Technology Agency, Atomic Energy Bureau, 22 September 1993)

1. The materials which come under this system include:
   All separated and recovered plutonium and highly enriched uranium being used for peaceful purposes
   Plutonium and highly enriched uranium that is produced through the dismantling of nuclear weapons, and which is being used for non-military purposes (hereinafter referred to as plutonium, etc.)
2. By unilaterally ascertaining the whereabouts and intended use of this plutonium, etc and publicly disclosing this information internationally, we hope to:
   i) Increase the transparency of activities pertaining to the peaceful use of plutonium, etc.
   ii) Verify internationally that the plutonium, etc. produced by dismantling nuclear weapons and being used for non-military purposes is being controlled appropriately
   It should also be known that this is not a system for inspecting whether plutonium is being converted to nuclear weapons.
3. The range of participating member countries in this international system, as a rule, covers countries in possession of plutonium, etc., and ensures impartiality between participating countries in the framework of this system.
4. Specific System Framework
   i) Participating member countries will, (a) record in this system the whereabouts of plutonium at each facility and regularly update this information based on subsequent activities, and (b) issue a projected plan of plutonium, etc. use, and report any surplus plutonium in storage control facilities that is not scheduled to be used immediately.
   ii) The taking out of surplus plutonium, etc. from the storage control facility is to be done by notification from the participating country.
   iii) The implementation of this system is basically administered by a committee of representatives from participating countries, but the executive director of this system realistically takes the role of the IAEA. The committee and executive director will be in charge of confirming the validity of reports and notifications made by participating countries.
5. The public disclosure of information called for in the summary of conditions for implementing this system should in no way interfere with the protection of nuclear materials, and should only be done to contribute to accomplishing the aforesaid purposes.

(3) Cost

One of the major criticisms voiced by many over the use of plutonium is cost. The argument is that it might be cheaper to use oil or uranium than it is to use plutonium which tends to be more expensive. We mentioned something about this earlier when we said that it was necessary to look at developing uses over the long term. A cost comparison study done by the OECD/NEA entitled, "Plutonium Recycling and Once Through" (Table 4), found no significant difference between the two types of processing. One of the things that has to be always kept in mind with regard to cost is that development is always moving ahead.

Table 4. OECD/NEA Nuclear Fuel Recycling Evaluation (1993)

1. Evaluation Preconditions
   We compared the annual cost of fuel recycling with the hypothetical case of a PWR nuclear reactor in the year 2000. As in the previous evaluation, two options were given, namely, the reprocessing option and the direct processing option.
   Using a method of calculation in which the values are adjusted to make them current, we compared each option using the latest data on reactor characteristics and fuel cycling costs.
2. Results
   Reference case nuclear fuel recycling cost
   Costs for both options are as follows:
   Reprocessing option: 6.23 mil/kWh
   Direct processing option: 5.46 mil/kWh
   The direct processing option was 12 percent cheaper than the reprocessing option, but this was because it was based on back-end cost. When front-end are included, it came out about the same.
   When we take into account the uncertainty of prices, we get a cost range between 5.17-7.06 for reprocessing, and a cost range of 4.28-6.30 for direct processing, which is not much of a difference.
   This difference cannot be simply dismissed when we consider that fuel recycling takes up 15-25 percent of the cost of generating electrical power. In making the choice whether to go with reprocessing or direct processing, one has to look not only at the economics of each option before deciding, but also at the energy policies of the country, the impact on the environment, and public acceptance.

4. Plutonium Supply and Demand Forecast

We mentioned it earlier in the context of nuclear non-proliferation, but policies regarding the use of plutonium in Japan must ensure and promote transparency. One of the key points in achieving that is the "principle that a country have no more plutonium than it needs." Since what is needed also includes the running inventory which a country maintains, supply and demand are not exactly equal every year. In Japan, the current supply of plutonium is less than demand, so Japan receives overseas shipments of plutonium which it had reprocessed abroad. In that sense, the aforementioned principle is fully attainable, and is also subsequently ensured through use in advanced nuclear reactors and light water reactors. In Table 5, we show a supply and demand forecast for plutonium up to the year 2000 that was taken out of a report published in August 1991 by a special subcommittee on nuclear fuel recycling in the Atomic Energy Commission of Japan.

Table 5. Projected Plutonium Use (Pu Supply/Demand to the Year 2010)

1. Plutonium Use		80-90 tons
(1) Projected Use by FBR and ATR		
Experimental "Joyo" FBR and prototype "Monju" FBR		12-13 tons
Demonstration FBRs and post-demonstration reactor		10-20 tons
Prototype "Fugen" ATR and demonstration reactor		10 tons
(2) Projected Use by Light Water Reactors		50 tons
Mid-1990s	2 reactors	(1/4 core)
Late 1990s	4 reactors	(1/3 core)
Beyond 2000	12 reactors	(1/3 core)
2. Plutonium Guarantees		85 tons
Tokai Reprocessing Plant		5 tons
Rokkasho Reprocessing Plant		50 tons
Overseas Reprocessing Plants		30 tons

(Atomic Energy Commission Special Committee on Nuclear Fuel Recycling)

5. Projected Plutonium Use in Japan

(1) Fast Breeder Reactor

The fast breeder reactor, which has a particularly high uranium utilization rate, is being positioned to become the main source of nuclear energy in Japan and the basis of plutonium usage. A plan is also in place toward putting FBRs into commercial use in the future.

The ability of this reactor to recycle transuranic elements also makes it better to cope with problems such as nuclear proliferation resistance and radioactive waste. That is why Japan is planning joint activities with the United States and France to perform R&D on fast breeder reactors.

(2) Advanced Thermal Reactors

In trying to build greater flexibility into the plutonium recycling system, the PNC is also working on an advanced thermal reactor that will have a wide degree of interchangability in terms of the nuclear fuel it uses.

(3) Use of Plutonium in Light Water Reactors (Pu-Thermal Reactor)

There are a number of countries around the world including France, Germany, and Switzerland which have practical experience using Pu-thermal reactors. Japan has also run some tests on these using small quantities of MOX fuel, but has found that Pu-thermal reactors might be better used as a passive "stopgap" measure until fast breeder reactors are commercially in operation. The lack of any significant difference in cost between MOX fuel and enriched uranium, and the nuclear non-proliferation principle of having no more plutonium than what is needed, have led us to a policy which favors the use of light water reactors as the principle means of generating nuclear energy.

6. Ensuring a Supply of Plutonium

(1) Light Water Reactor Reprocessing

In April 1993, construction began on the Rokkasho Reprocessing Plant. The plant is scheduled to go into operation sometime after the year 2000 and will be the primary supplier of plutonium in the future.

The Tokai Reprocessing Plant has the distinction of being the second light water reactor reprocessing plant in the world after one in France. Plans call for recovered plutonium from this plant to be used in R&D on fast breeder and advanced thermal reactors. When the Rokkasho Reprocessing Plant goes into operation, the main role of this plant will shift toward R&D on future reprocessing.

Overseas reprocessing is only an interim measure until the Rokkasho Reprocessing Plant is up in operation, thus concluding any future overseas reprocessing agreements has to be weighed carefully even with the experience gained in shipping plutonium overseas.

7. About Mixed Oxide Fuel (MOX)

(1) Manufacturing System for MOX Fuel

The PNC has the role of manufacturing MOX fuel for fast breeder and advanced thermal reactors.

The manufacture of MOX fuel for light water reactors is currently being researched by the electric companies. Once the Rokkasho Reprocessing Plant is in operation, it will be necessary to commercialize this manufacturing process within a specific period of time.

As far as recovered plutonium by overseas reprocessing, it is appropriate as things stand now that MOX fuel processing be done overseas, and studies are being done by electric companies to make this happen.

(2) Reprocessing of MOX Fuel

The PNC is currently doing R&D on the reprocessing of fast breeder reactor fuel, and is scheduled to start construction on a hot engineering-scale recycling equipment testing facility (RETF) that will likely be in operation by the year 2000.

The reprocessing of ATR fuel and LWR MOX fuel can basically be done with LWR fuel technology, so the necessary information is being collected to make this into a future project.

8. Return Shipment of Plutonium

A total of 30 tons of plutonium is scheduled to be returned from abroad. As we mentioned already, not only are there no specific ways of using plutonium in Japan, but the plutonium being brought back to Japan belongs to Japan in the first place, so in that sense Japan will never have more plutonium than what is needed.

As things stand now, plutonium is basically transported back to Japan by ship. Japan is determined to keep satisfying international safety standards while faithfully honoring and implementing bilateral nuclear energy agreements and treaties regarding the security of nuclear materials. The next shipment of plutonium is scheduled in arrive in three to five years at the earliest.

Plutonium Usage for Advanced Reactors

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[Article by K. Ito and Y. Hayamizu, Power Reactor and Nuclear Fuel Development Corp (PNC), Power Reactor Development Division]

[Excerpts] Countries poor in energy resources are being forced to switch their main source of energy from fossil fuels to nuclear energy in order to ensure long-term stable supplies of energy. Being one of those countries, Japan is busy trying to develop advanced reactors (fast reactors and advanced thermal reactors) which have the unique capability of burning both uranium and plutonium obtained from light water reactors, and trying to perfect a uranium-plutonium nuclear fuel recycling system that makes the most effective use of uranium resources.

Since fast reactors can be made into breeder reactors that produce additional nuclear fuel above and beyond that used to generate energy, the commercialization of fast reactors will mark the official use of plutonium and will dramatically increase the amount of uranium resources used.

Advanced thermal reactors have great flexibility in terms of the use of plutonium, recovered uranium, and depleted uranium, and thereby significantly reduce the amount of natural uranium required and amount of uranium enrichment.

In what follows, we will describe the use of plutonium in fast reactors and advanced thermal reactors.

Part 1. Plutonium Use in Fast Reactors

1. Properties of Plutonium Use in Fast Reactors

(1) Breeding Properties

The most important characteristic of a fast reactor is that it can produce, or "breed," large quantities of new nuclear fuel from nuclear fuel not burned in a nuclear reactor while the reactor is running.

Theoretically, breeding takes place when a single neutron is absorbed by the fuel and more than two fission neutrons are produced during nuclear fission, but optimum breeding takes place using plutonium in a fast reactor because plutonium will use fast neutrons to produce the greatest number of neutrons.[1] The core structure of a breeder reactor consists of a reactor core unit at the center loaded with a mixture of uranium-plutonium fuel. Blanket fuel is placed around the top, bottom, and radial periphery of the core, which is then loaded with either depleted or natural uranium.

(2) Np, Am and Other Minor Actinide Fuels

In fast neutron regions, the fission cross section of minor actinides (hereinafter MA) such as ^{237}Np and ^{241}Am expands dramatically. That is why MAs can be used effectively in fast reactors.

Fast reactors, consequently, have the advantage of being able to recycle fuel many times over without having to upgrade plutonium, which is one of the problems with fuel recycling by light water reactors.[2] This effectively cuts down the amount of MA in waste from high-speed reactor fuel recycling, and makes it easier to treat and dispose of waste.

2. Current Status of Fast Reactor Development

[passage omitted]

(2) Current Status of Fast Reactor Development in Japan

The development of fast reactors in Japan has been proceeding step by step from the experimental reactor stage, to the prototype reactor, and to the demonstration reactor, with the goal being to put fast reactors into commercial use between the years 2020 to 2030, based on a basic policy of using domestic technologies to develop the breeder reactor rather than importing technologies from abroad. The developmental work up to the prototype reactor stage was basically done by the Power Reactor and Nuclear Fuel Development Corporation (hereinafter PNC) with the cooperation of various industries.

(i) Experimental "Joyo" Fast Breeder Reactor

The experimental "Joyo" fast reactor, which was the first stage in the development of fast reactors in Japan, was designed in 1962 by the Japan Atomic Energy Research Institute (hereinafter referred to as JAERI). The PNC continued with design and construction after that and took the reactor critical in April 1977. In Table 1, we show the main specifications of the experimental "Joyo" fast breeder reactors.

The experimental "Joyo" fast breeder reactor achieved a rated thermal output of 50,000 kW and 75,000 kW with the MK-I core (breeder core), and in August 1983, started using the MK-II core (irradiation core), wherein it achieved a rated thermal output of 100,000 kW. Various tests were conducted at that time on the irradiation of fuel and material, and in 1984, plutonium was recovered from the spent fuel of the experimental "Joyo" FBR, and by refueling the reactor again with one pin of new fuel, the loop of the nuclear fuel cycle based on the fast breeder reactor was successfully closed.

Table 1. Main Specifications of Fast Reactors in Japan

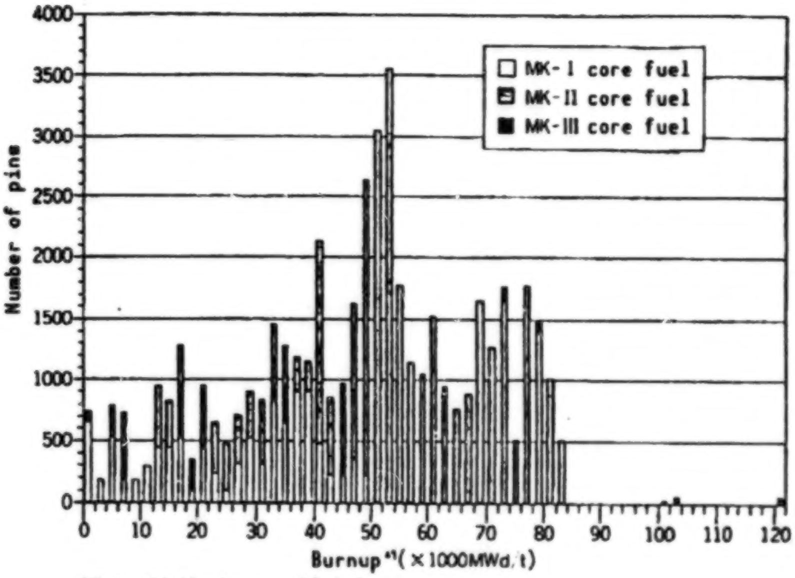
	Experimental "Joyo" FBR		Prototype "Monju"	FBR No. 1 Experimental Reactor (preliminary conceptual design)
	Fast breeder core (MK-I)	Fast Breeder Core (MK-II)		
1. Output Thermal output Electrical output	75MW —	100MW —	714 MW 280 MW	1600 MW 670 MW
2. Thermal Efficiency	—	—	39%	42%
3. Reactor core Height Diameter Capacity Cooling systems (primary/secondary) Cooling systems (inlet/outlet)	600mm 800mm 304 l 2/2 370°C/470°C	550mm 730mm 250 l 2/2 370°C/500°C	930mm 1180mm 1340 l 3/3 397°C/529°C	1000mm 2990mm — 3/3 395°C/550°C
4. Fuel Pins/assemblies Assemblies U loading Pu loading Pu enrichment Pu.f/(PuO_2-UO_2) Blanket U Breeding ratio Burnup (average assembly)	91 82 177 kg U^{235} 166 kg 18 w/o 6.2t 1.01 42,000 MWd/t	127 67 100 kg U^{235} 220 kg 20 w/o — — 75,000 MWd/t	169 198 4.5t (initial) 1.4t (initial) Inner 18 w/o Outer 21 w/o 17.5t 1.2 80,000 MWd/t	217 295 — — — — — 1.2 (1.05 w/o radial blanket) 90,000 MWd/t
5. Control rods	6 (4-safety, 2-regulating)		19 (backup, shutdown)	30 (backup, shutdown)
6. Miscellaneous Reactor vessel model Containment vessel system Containment vessel height Fuel exchange method Steam condition	Upper flange cylindrical Semi-double-layered system 54.4m Double rotation-type		Upper flange cylindrical Semi-double-layered system 79m Rotating fixed arm 484°C 127 kg/cm^2	Loop slab integrated cylindrical Positive-pressure type 65m Double rotation 495°C 169 kg/cm^2

As irradiation tests are being done on "Monju," FBR fuel and data collected on high-burnup fuel, operation support and equipment monitoring systems are being developed and those results are being used in the operating maintenance system of the "Monju" reactor.

Irradiation tests are also being done on the "Joyo." An irradiation test is being conducted in joint cooperation between France and Japan to improve the performance of fuel material and structural material used in fast breeder reactors, and another test is being done in conjunction with a university on fusion reactor materials and new element materials.

The experimental "Joyo" FBR has been continuously used in this way since being put into operation, and during that time has irradiated approximately 49,000 pins of U-Pu mixed oxide fuel (hereinafter referred to as MOX fuel), steadily collecting in the process the technical information and operating experience needed to develop a prototype FBR reactor (Figure 1).

In order to do a broader study of irradiation properties of the experimental "Joyo" FBR, which includes developing new fuels such as nitrous oxide fuel and minor actinides, both of which are envisaged in the broader context of FBR development, a plan is underway for aJoyo MK-III core that will have far better irradiation



(Note 1) Maximum pellet burnup
Burnup histograph up to 26 March 1993

Figure 1. Demonstration "Joyo" FBR Fuel Irradiation Performance

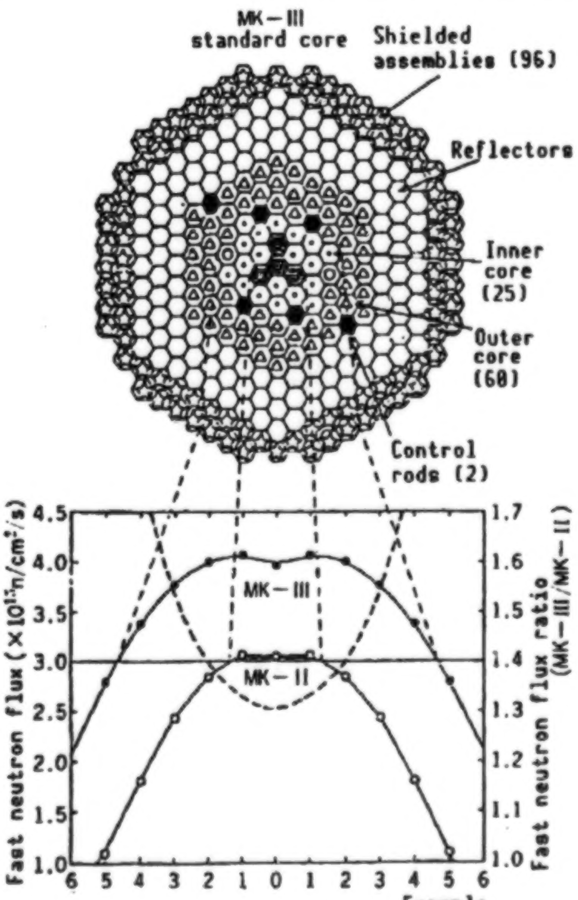


Figure 2. Comparing Fast Neutron Flux of MK-II and MK-III Core

properties. The new core is projected to increase thermal output to 140,000 kW, double the amount of irradiation, and have twice the irradiation space (Figure 2).[4]

(ii) Prototype "Monju" Fast Breeder Reactor

The purpose of developing the prototype "Monju" fast breeder reactor (electrical output: 280,000 kW) has been to ascertain the performance and reliability of a fast breeder reactor as a power generation plant through the design, construction, and operation of that reactor, and also to obtain the necessary data for studies and evaluations relating to cost.

To accomplish these objectives, the "Monju" FBR has been built entirely with domestic technology, much of it from the R&D and experience acquired in designing, building, and operating the experimental "Joyo" FBR. Much of the technical resources of the country have been focused to develop the MOX fuel and MOX fuel manufacturing technology.

Construction started on the "Monju" FBR in October 1985, and by April 1991 the installation of equipment was complete. Comprehensive testing was performed between May 1991 and December 1992 to verify functional design of systems and equipment. The "Monju" is currently undergoing a criticality test while fuel loading as part of the next stage of performance testing. With the criticality test, the PNC hopes to verify minimum criticality by April 1994. After a minimum critical core is configured and minimum critical core characteristics are verified including the measurement of control rods and peripheral fuel, the remaining core fuel will be added and initial fueling will be complete.

That test is followed by a physical test of the reactor done at extremely low output for the purpose of studying the reactivity characteristics of the control rods, to measure and evaluate output distribution, and to evaluate shielding around the periphery of the reactor. This test in turn is followed by a nuclear heating test in which the systems undergo nuclear heating, and the water/steam and turbine systems are started, tuned and initially linked with the electric power system. An output test is performed to verify the operating, control, and transient characteristics of the plant for 40 percent, 75 percent, and 100 percent output phases (Figure 3).

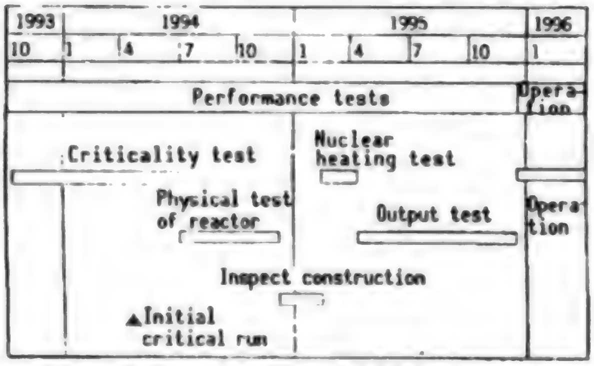


Figure 3. Prototype "Monju" FBR Performance Test Plan

Radiation shielding and sodium chemical analysis tests are done continuously in conjunction with those tests as output rises, giving a total of 130 items to be completed in the performance tests by December 1995.

(iii) Demonstration Reactor

The design, construction, and operation of the demonstration reactor, the successor to the prototype reactor, has for the most part been consigned to a joint venture formed between the electric power companies and PNC. The related R&D is consigned to the power companies, the PNC, and other related organizations, indicative of a national policy toward trying to allocate the research and development work most effectively.

In 1985, the electric power companies contracted much of the design, construction, and operation of the demonstration reactor to the Japan Nuclear Power Co., Ltd. (hereinafter referred to as JNP). JNP conducted most of the R&D related to element technologies and main equipment, and along with that handled much of the non-specified design research on the reactor model which included theoretical design research, innovative technology research, and study of systems involving innovative technologies. In 1990, using the results gained from this R&D, JNP decided to go ahead with preliminary conceptual design research based on a 60,000 kW top-entry loop-type reactor. It has been conducting verification tests on plant design and technical sophistication, and at present, is in the process of finalizing the main specifications of the demonstration reactor. The main specifications of the demonstration reactor are given in Table 1.[5]

In order to coordinate the R&D supporting the design and construction of the demonstration reactor, a fast breeder reactor R&D steering committee was set up in 1986 with members of the committee coming from the PNC, JNP, JAERI, and the Central Research Institute of the Electric Power Industry (CRIEPI). The committee has been holding discussions and examining the specifics of post-demonstration reactor R&D according to a national FBR development policy.

(3) Future Development

(I) Developing Technology for Commercializing the Fast Breeder Reactor

In order to achieve the goal of putting fast breeder reactors into commercial use between 2020-2030, the PNC intends to use the demonstration reactor until the year 2010 to demonstrate safety and reliability, verify cost, and gain a better understanding of technical feasibility.

The goals that must be achieved in order to commercialize the fast breeder reactor are summarized below:

1. Make the FBR safer than the LWR by taking advantage of the inherent safety of fast reactors, and develop a more advanced safety assessment method.
2. Make the FBR highly reliable with excellent durability by making smart use of technology and the experience gained from advanced reactors.
3. Lengthen the operating cycle of the FBR by improving the burnup of fuel in the reactor.
4. Make the FBR cost-competitive with a LWR by achieving compactability and better performance of equipment and instruments through technical breakthroughs.

Some of the suggestions put forth to achieve these goals include improving fuel and core performance, improving the thermal efficiency of the plant, upgrading the equipment that make up the plant, introducing new technologies, and building upon the inherent safety of fast reactors. Figure 4 is a conceptual image of a plant in which the technologies now undergoing R&D which are considered key to achieving these goals have been incorporated into the design of a plant.

We can see from Figure 5, which is a transitional drawing showing plant construction costs as new developmental items are introduced, that the FBR is cost-competitive with the LWR right from the start, and that it would be prudent to move quickly in resolving the remaining problems.

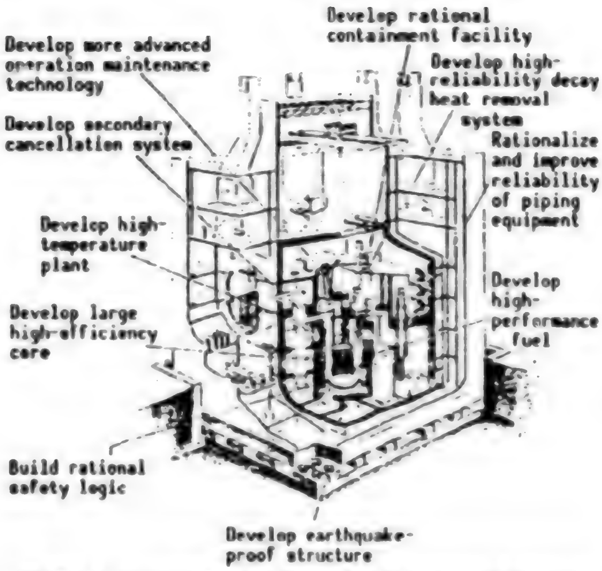


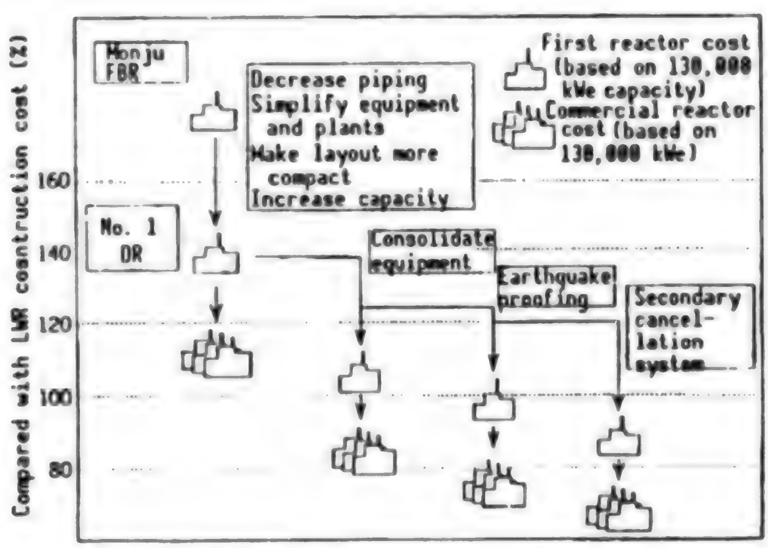
Figure 4. Developmental Goals for Commercialization

improve that to 90,000 MWd/t by making the fuel cladding from an upgraded austenite steel that is resistant to high temperatures.

The burnup goal for commercial FBR use is estimated to be about 200,000 MWd/t, so what the PNC is trying to do is to develop an oxide-dispersed ferrite steel having not only low swelling properties but is also resistant to high temperatures.

One of the ways of stopping reactor cores from growing further in size as reactor output increases, and of making the cores more compact, is by increasing the maximum linear output of fuel. The linear output limit of fuel elements has to do with preventing fusion, so what the PNC is trying to do is to gain a more precise understanding of this phenomenon and improve the ways in which this is analyzed and evaluated, and at the same time collect and organize data on fuel irradiation.

In doing this, the PNC has had to try to create better coordination and compatibility between fuel manufacturing and reprocessing technologies.



DR: Demonstration reactor

Figure 5. FBR Plant Construction Cost

In the sections below, we discuss problems associated with improving fuel and reactor core performance:

(a) Development of High-Performance MOX Fuel

At the commercial level, the key to a longer operating cycle and cheaper fuel cycle is better fuel burnup. This calls for a fuel cladding material that is resistant to high temperature and radiation and exhibits good compatibility with sodium.

The burnup performance of the "Monju" reactor is about 80,000 MWd/t, but it is technically feasible to

(b) Development of High-Performance Reactor Core

In order to achieve lower operating cost at the commercial level by a longer operating cycle, the PNC has been trying to flatten output and improve performance of the reactor core, and in so doing has found that it needs to come up with an optimum core design that will translate into a better fuel burnup. That means it needs to gain a more precise understanding of the characteristics pertaining to various core systems, and develop more advanced analysis

and evaluation methods and build a larger data base related to nuclear design, shielding design, and core heat flow design.

(ii) R&D To Seek Fast Reactor Possibilities

(a) Advanced Fuels

Nitrous oxide fuel takes advantage of the high density and excellent thermal conductivity of heavy metals to achieve high fuel burnup (approximately 12,000 MWt/d and breeding ratio (around 1.2). For that reason, a study is being done on a nitrous oxide fueled core with an enhanced passive safety characteristic (Table 2).

Table 2. Specifications of Nitrous Oxide Fuel Reactor

Reactor thermal output	2,600 MW
Operating cycle length	365 days
Fuel exchange batches	5
Rated maximum linear output	450 W/cm
Average linear output	260 W/cm
Core height	70 cm
Core equivalent diameter	369 cm
Axial blanket thickness	40 cm
Internal blanket equivalent diameter	100 cm
Radial blanket thickness	33 cm
Average core burnup	125,000 MWd/t
Breeding ratio	1.19
Reactivity	2.6% $\Delta k/kk'$
Pu enrichment (inner/outer core)	15.7/19.0(wt%)
Peak velocity fluence	4.5×10^{23}
Coolant temperature coefficient	0.007C/°C

If we use a nitrous oxide fuel reactor with the same linear fuel output as a conventional oxide fuel reactor, we can lower the average fuel temperature during normal operation. In doing this, we have found even during unprotected loss of flow (ULOF) that the reactivity of the entire core acts negatively due to the passive safety characteristic and the cooling material never reaches boiling (Figure 6). Even in cases of excess reactor output, we have found there to be a large margin of safety when it comes to the melting of fuel.

On the other hand, because the temperature of nitrous oxide fuel is lower than that of oxide fuels, there is a possibility without the creep down characteristic of a deterioration in the mechanical interaction between the fuel and cladding. That is the reason for the two-pin irradiation test being planned in 1994 with the experimental "Joyo" FBR. A study is also being undertaken to evaluate enrichment cost and to identify the nuclear properties of ^{15}N, which is needed to manufacture nitrous oxide fuel.[7]

(b) Plutonium Burning Cores

A study is being done on a fast reactor design that will help cope with the fluctuations in plutonium supply and

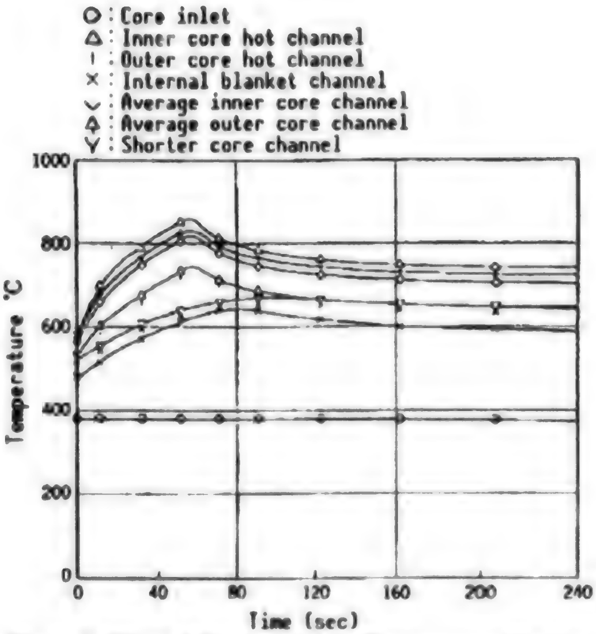


Figure 6. Channel Output Coolant Temperature During ULOF of Nitrous Oxide Fuel Core

demand. The reactor will have a higher degree of safety in that it will not be limited to breeding, but will use plutonium having longer fuel life. To find newer ways of using plutonium, studies are also being done on high plutonium rich reactors and uranium-free reactors. When we compare these reactors with the core characteristics of conventional fast reactors, we find them to be inadequate in terms of safety as it pertains to excess reactivity, the doppler coefficient, and the percentage of delayed neutrons.

(c) Minor Actinide (MA) Burning Cores

The following research is being done to develop a technology for using minor actinides and other long-lived radioactive nuclide as fuel for fast reactors.

1. Evaluating Characteristics of MA Fuel Added Cores

Two methods are currently under consideration for fueling a reactor core with MA, namely, using a homogenous core in which the MA is added to the overall core fuel, and a heterogeneous core in which the MA is added to specific fuel assemblies and these assemblies are placed together or dispersed throughout the core. The major advantage of using a heterogeneous core is the cost, in the sense that it requires the manufacture of fewer MA fuel assemblies, and can cope more effectively with fluctuations in the supply of MAs. On the other hand, annihilating a large quantity of MA requires much more MA to be added to specific fuel assemblies, and this will negatively affect the physical properties of the fuel such as melting point and thermal conductivity.

An analysis of the characteristics of each of the two types of 100,000 kW MA burning cores mentioned above is now underway (Table 3). According to the research so far, allowable output (approximately 430 W/cm) is somewhat more for fuels in which MA is not added to a heterogeneous core, but by optimizing the fuel pin diameter and the fueling pattern it will be possible to correct this, so we believe that there is a future for both cores, the homogeneous and heterogeneous core, as MA fuel burning cores.

Table 3. Comparison of MA Loaded Core Characteristics

Standard core homogenous MA	Heterogeneous MA	Without MA loaded core (5%)	Loaded MA core (50% added to 39 units)
Pu enrichment (wt%) (inner/outer/core)	15.3/19.3	16.2/19.6	15.3/19.3
Max linear output (W/cm)	420	431	439/310 (normal/MA added)
Burnup reactivity loss (%Δk/kk')	3.3	1.9	1.9
Primary control rod loss (%Δk/kk')	1.67	1.67	1.50
Doppler coefficient (Tdk/dT)	-10.5×10^{-3}	-7.1×10^{-3}	-7.4×10^{-3}
Coolant temperature coefficient (Δk/kk'/100% density change)	-1.73×10^{-2}	-2.50×10^{-2}	-2.60×10^{-2}
MA annihilation (kg/cycle)	—	183	189
MA annihilation (%)	—	11.7	12.1

It has also been found, regarding material balance, that it will be possible to annihilate approximately 150 kg of MA per year, which is equal to the amount of MA produced annually by six 100,000 kW light water reactors.[8]

(ii) Nuclear Data Verification of MA Nuclide

An evaluation is currently being done with the experimental "Joyo" FBR on the annihilation rate of ^{241}Am in which one fuel pin containing about 7% ^{241}Am is irradiated to 40,000 MWd/t. That will be followed in 1994 with irradiation tests on Np, Am, and Cm samples which will be done by the experimental "Joyo" FBR. After those samples, plans call for a gamma spectrometry, instrument analysis, and precise assessment of nuclide transformation efficiency and nuclear cross sections of MAs.

Another test has been started to verify the accuracy of fission cross sections of MA nuclide by inserting fission chambers deposited with small quantities of MA nuclide into the irradiation holes of a fast neutron reactor at Tokyo University.

(iii) Irradiation Behavior of MA Additive Fuel

Once a manufacturing technology for MA-added fuel has been developed, plans call for an evaluation of irradiation behavior with the experimental "Joyo" FBR on fuel composition (percentages of Np, Am, and Cm), linear output, and short-pin fueling as part of the full-scale irradiation tests on MA fuel pins. A facility is now being built to manufacture MA-added fuel.

Part 2. Plutonium Usage in Advanced Thermal Reactors

Because advanced thermal reactors (ATR) exhibit both good fuel utilization and excellent flexibility when it comes to plutonium usage, the government has decided to develop these reactors with Japanese technology exclusively in accord with a policy for them to play a supporting role in putting fast breeder reactors into commercial use. The prototype "Fugen" ATR has been running smoothly ever since it was put into operation in 1979, and during that time it has burned approximately 500 pins of MOX fuel and has demonstrated the ability to use plutonium. The Dengen Kaihatsu Co, Ltd., the main builders of the next demonstration ATR, are busy working on plans to build a reactor in Aomori Prefecture.

1. Main Features of Advanced Thermal Reactors (ATR)

In ATRs, heavy water is used as a moderator and the fuel and moderator are maintained separately, thereby producing less loss of neutrons in the slowing-down process than light water reactors and giving ATRs the following advantages regarding fuel use.

(1) Use of Reprocessed Fuels

The difficulty with reprocessing and reusing spent fuel in an LWR is the detrimental effect that higher plutonium isotopes, americium, and U-236 have on reactivity. These effects are far less harmful in an ATR, which enables ATRs to handle a greater variety of fuels without having to differentiate between types of fuel.

In Figure 7, we show the effect that uniform fissionable plutonium isotope composition has on reactivity. As burnup of fuel increases, the Puf/Pu tends to grow smaller, but with the ATR we can see that this effect is negligible.

(2) Effectively Burns Weaker Fuels

The amount of uranium or plutonium needed to get the same amount of heat is about two-thirds that of an LWR.

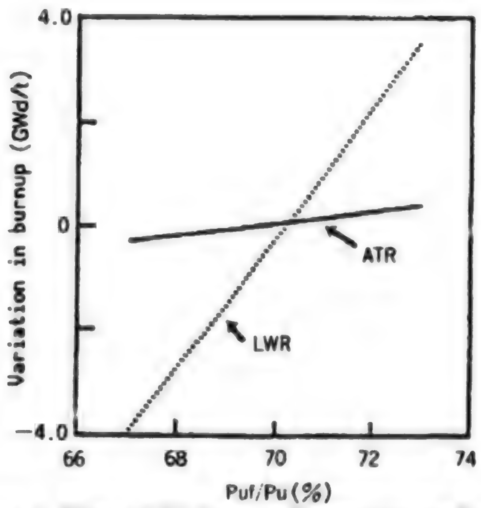


Figure 7. Effect of PU Isotopic Composition on Burnup

2. Specifics of ATR Development

In May 1966, the decision to develop and build an advanced thermal reactor became a national project.

The prototype "Fugen" ATR (Photo 1 [not reproduced]) is Japan's first attempt to actually develop an advanced power reactor, and was built entirely with Japanese expertise, which meant that the entire design, fabrication, construction and operation were done with domestic technologies. That meant that the PNC had to build a full-scale test model at the O-Arai Engineering Center with which to conduct durability and safety tests of equipment, the results of which were later used in the design and fabrication of the prototype "Fugen" ATR.

Construction of the "Fugen" ATR began in December 1970 and it went into full-scale operation on 20 March 1979. Over the course of more than 15 years of operation, the "Fugen" ATR has demonstrated reliability in its design, equipment, and use of MOX fuel.

3. Actual Results of Plutonium Usage

(1) Quantity of Plutonium Used

By December 1992, the reactor had been loaded with 507 pins of MOX fuel and 464 pins of uranium fuel. The 507 pins of MOX fuel loaded contained 11,000 kg of plutonium, of which approximately 800 kg was fissionable plutonium. This performance was better than any other type of thermal reactor in the world.

In Figure 8, we show the actual results of MOX fuel use in thermal reactors around the world.

(2) Flexibility in Percentage of MOX Fuel

The percentage of MOX fuel within the core can go as high as 100 percent, but it varies depending on the availability of MOX fuel. It starts at approximately 43 percent when the core is filled initially and drops to about 34 percent by the fourth cycle, but fuel is then gradually added, and by the 16th cycle it reaches its highest point at 72 percent. Even though the percentage of MOX fuel varies widely, it affects neither core nor control characteristics to any great degree, and demonstrates the abundant flexibility of the ATR when it comes to using MOX fuel.

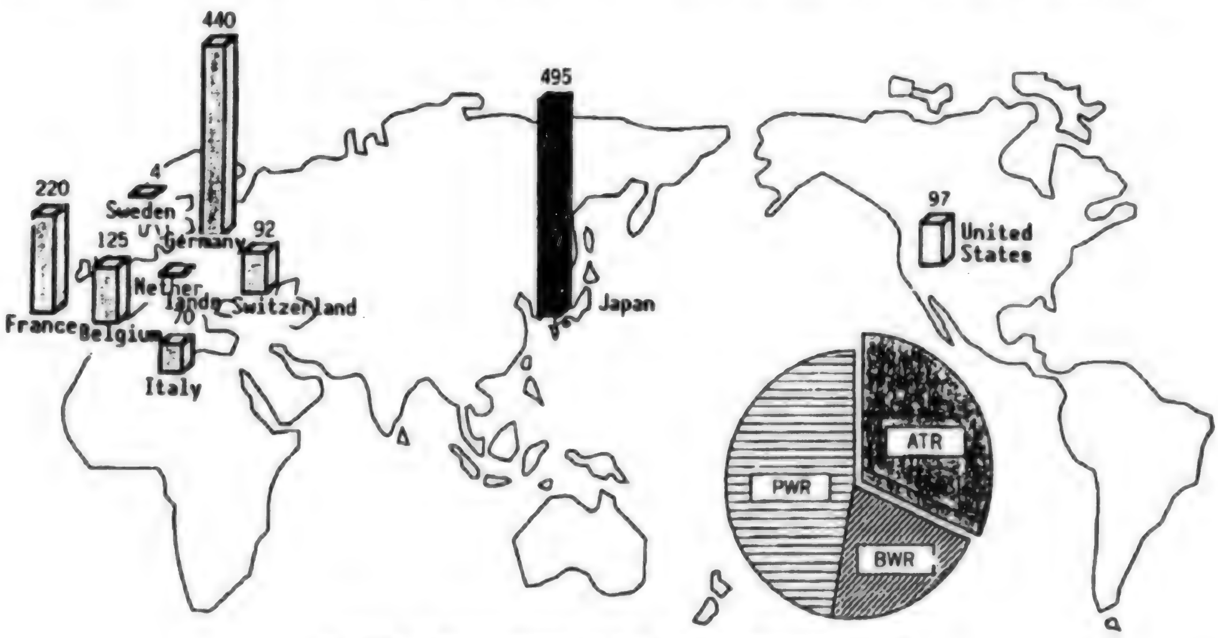


Figure 8. Global MOX Fuel Use (as of December 1991)

(3) Use of Higher Plutonium Isotopes

The plutonium recovered by reprocessing 34 pins of spent fuel from the "Fugen" ATR was used in four pins of MOX fuel loaded into the core in June 1988. Though small, this was proof, nevertheless, that recovered plutonium could be easily used without any major changes to core or control characteristics. As far as the isotopic composition of recovered plutonium, 70-80 percent of the plutonium recovered from the spent fuel of an LWR is fissionable plutonium (^{239}Pu and ^{241}Pu), compared to 60-65 percent of that recovered from spent MOX fuel of the "Fugen" ATR. This is due to an increase in higher plutonium isotopes as plutonium recycling continues. This performance demonstrates that the ATR can adequately deal with higher plutonium isotopes either resulting from better LWR burnup or resulting from the recycling of Pu-thermal fuel.

4. Issues Remaining in the Future

(1) Advanced Technology for Use of Plutonium in Demonstration Reactor

The company in charge of building the demonstration reactor, Dengen Kaihatsu, has completed the basic design and is making preparations to start building the reactor in the town of Oma at the tip of the Shimokita Peninsula in Aomori Prefecture with the goal of having the plant in operation by March 2001. The MOX fuel which it will develop for the demonstration reactor will have a 30 percent higher channel output and 90 percent higher burnup than the "Fugen" ATR, and will be used throughout the entire core.

(2) Commercial Development

The following strategy promoting the use of plutonium was set forth in 1987 in a long-range plan on developing uses of nuclear energy.

(i) It will be necessary to quickly demonstrate plutonium recycling on a scale equal to both an LWR and an ATR, and to try to establish a base for plutonium usage including reprocessing and MOX fuel manufacture and to try to improve maintenance of the peripheral environment and overall cost of the nuclear fuel cycle.

(ii) The advanced thermal reactor is a heavy water reactor which has characteristics particularly suited to the use of plutonium, recovered uranium, and depleted uranium, and also has wide flexibility in terms of nuclear fuel including the ability to fill the core entirely with MOX fuel. In view of trying to establish a technical base consisting exclusively of Japanese technology, it will be necessary to try to improve cost with an eye toward commercialization and make a more advanced heavy water reactor technology through further development of the reactor.

Given the above strategy, the following work is being done to develop a basic technology for the ATR:

(1) Simplifying the reactor core based on average channel output within the pressure lines

(2) Lowering fuel cost by improving burnup of fuel

(3) Doing research to improve the safety of the reactor

(4) Doing research on diversified fuels

Bibliography

(1) Yasu: "Fast Breeder Reactors," (same study), October 1982.

(2) Wakayabashi, Ikegami: "Characteristics of an LMFBR Core Loaded with Minor Actinide and Rare-Earth Containing Fuels," PNC TN 9410 93-100.

(3) P. Coulon and L. Martin: "Phoenix Operation Experience Fuel Management and Fuel Cycle," FR '91, International Conference on Fast Reactors and Related Fuel Cycles, October 1991, Kyoto, Japan.

(4) Yanagizawa: "Research Reactor of Power Reactor and Nuclear Fuel Development Corporation," *Genshiryoku Kogyo*, Vol 39 No 11, 1993, pp 18-23.

(5) Miura, et al: "Profile of Top Entry-Type FBR Demonstration Reactor," *Nihon Genshiryoku Gakkai-Shi*, Vol 35 No 4, pp 268-279 (1993).

(6) Nagata, et al: "Design Research on Large-Scale Fast Breeder Reactor," PNC TN 1340, 92-002 *Donen Giho*, No 82, Jun 1992, pp 19-37.

(7) "PNC: Current Status of Fast Breeder Reactor R&D," PNC TN 1410 93-004, February 1993.

(8) Wakabayashi, et al: "Current Status of Nuclear Transformation Research Using Fast Reactors and High-Speed Instruments," PNC TN 1340 92-002, No 82 *Donen Giho*, June 1992, pp 96-103.

(9) PNC: "Donen Giho O-Arai Engineering Center," PNC TN 1340 90-001, March 1990.

(10) PNC: "Summary of Prototype "Fugen" ATR Technical Research Findings," PNC TN 1410 91-054, Aug 1991.

Plutonium Usage for Light Water Reactors

94FE0521C Tokyo GENSHIRYOKU KOGYO in Japanese Jan 94 pp 26-30

[Article by S. Muto, Tokyo Electric Power Co, Ltd., Nuclear Energy Division]

[Excerpts]

1. Significance of Plutonium Usage in Japan

The basic line of reasoning which Japan has followed from the very beginning regarding developing uses for nuclear energy has been that spent fuel would be reprocessed and the plutonium obtained thereof would be reused in order to form what is called "fuel recycling." The thinking behind this is that in a resource poor country such as Japan uranium resources could be used more efficiently, and there would be less threat to the environment due to recycling and this would make nuclear energy a much safer and more economical energy resource over the long range and would be easier to manage radioactive waste.

Fast breeder reactors have a high utilization rate for uranium resources, and this forms the basis of "fuel recycling." In a global context, the generation of nuclear energy is, for the most part, done by light water reactors, and we believe that it will be quite some time before fast breeder reactors are put into commercial use, so we find no problem right now using the plutonium obtained by reprocessing with light water reactors (Pu-thermal use). Doing this enables us to employ reprocessing in making U-Pu mixed nuclear fuel (hereinafter referred to as mixed nuclear fuel), and helps us ready a nuclear recycling system for the future, preventing unnecessary stockpiling of plutonium and making Japan more energy self-sufficient.

2. Pu-Thermal Usage Plan in Japan

Based on the line of reasoning stated above, Japan has decided to continue with Pu-thermal use at an appropriate scale, but has also decided at some time in the future to expand the scale based on results from demonstration tests done on a smaller scale.

The plutonium needed to do that will be supplied by the PNC Tokai Reprocessing Plant, the JNF (Japan Nuclear Fuel, Ltd.) Rokkasho Reprocessing Plant, and that contracted to overseas reprocessing plants in the United Kingdom and France. The fabrication of fuel, on the other hand, is being done overseas where plutonium recovered from overseas reprocessing plants is made into mixed nuclear fuel and shipped overseas to Japan, but a plan is also in the works for mixed nuclear fuel to be done in Japan by the privately run Rokkasho Reprocessing Plant. Another important feature of this is that a technology is being developed to reprocess the spent mixed nuclear fuel generated by this recycling.

[passages omitted]

4. Pu-Thermal Performance To Date

(1) Plutonium Usage in Japanese Light Water Reactors

The Japan Atomic Energy's No. 1 Tsuruga reactor and Kansai Electric Power's No. 1 Mihama reactor have completed loading and irradiating a small quantity of mixed nuclear fuel in the cores of their respective reactors without any problem so far. In Figures 4 and 5, we show the respective layouts of the inner assembly fuel rods in each reactor. The Tsuruga design is what is known as an island-type and the Mihama is a discreet-type. In Tables 1 and 2, we list the design specifications of the assemblies. In the Tsuruga reactor two assemblies were irradiated through three cycles from July 1986 to January 1990, and in the Mihama reactor four assemblies were irradiated through three cycles from April 1991 to December 1991. The plutonium portion of the assembly at Tsuruga was manufactured by the PNC and the uranium portion by Japan Nuclear Fuel, whereas the plutonium portion of the assembly at Mihama was manufactured by Westinghouse in the United States, even though part of the assemblies contained pellets made by the PNC. After the irradiation of both assemblies, post-irradiation tests were conducted in a hot laboratory.

Table 1. Design Specifications of No. 1 Tsuruga Reactor Mixed Nuclear Fuel

	MOX fuel rod	Uranium fuel rod
Pellet diameter (cm)	1.03	1.03
Pellet length (cm)	1.0	1.0
Pellet density (% TD)	95	95
Pellet material	UO_2-PuO_2	UO_2 UO_2-Gd_2O_3
Outer cladding diameter (cm)	1.23	1.23
Cladding thickness (mm)	0.86	0.86
Cladding material	Zircaloy-2 (recrystallized annealed material)	Zircaloy-2
Effective fuel rod length (m)	3.66	3.66
Pellet cladding spacing (mm)	0.24	0.24
Maximum linear output density (kW/m)	44.0	44.0
Maximum pellet temperature (at design linear output density) (°C)	1470	1850
Maximum external cladding temperature (°C)	390	
Fuel assembly length (m)	4.35	4.35
Enrichment (assembly average)* (wt%)	2.9	
Outer water rod diameter (cm)	1.50	

Note: * $(^{235}U + ^{239}Pu + ^{241}Pu)/(U + Pu)$

Table 2. Design Specifications of No. 2 Mihama Reactor Mixed Nuclear Fuel

	MOX fuel rod	Uranium fuel rod
Pellet diameter (cm)		0.929
Pellet length (cm)		1.25
Pellet density (% TD)		92
Pellet material		UO_2-PuO_2
Outer cladding diameter (cm)		1.072
Cladding thickness (mm)		0.62
Cladding material		Zircaloy-4
Total fuel rod length (m)		3.21
Pellet cladding spacing (mm)		0.21
Maximum linear output density (kW/m)		59.1
Maximum pellet temperature (at maximum linear output density) (°C)		2500
Maximum external cladding temperature (°C)		350
Fuel assembly length (m)		3.5
Enrichment (assembly average)* (wt%)		3.8

Note: *: $(^{235}U + ^{239}Pu + ^{241}Pu)/(U + Pu)$

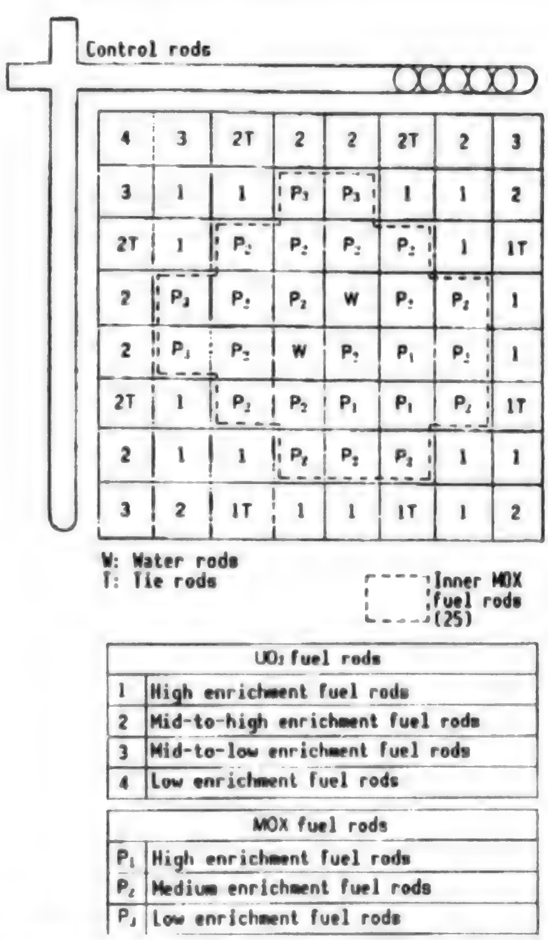


Figure 4. No. 1 Tsuruga Reactor Mixed Nuclear Fuel Design

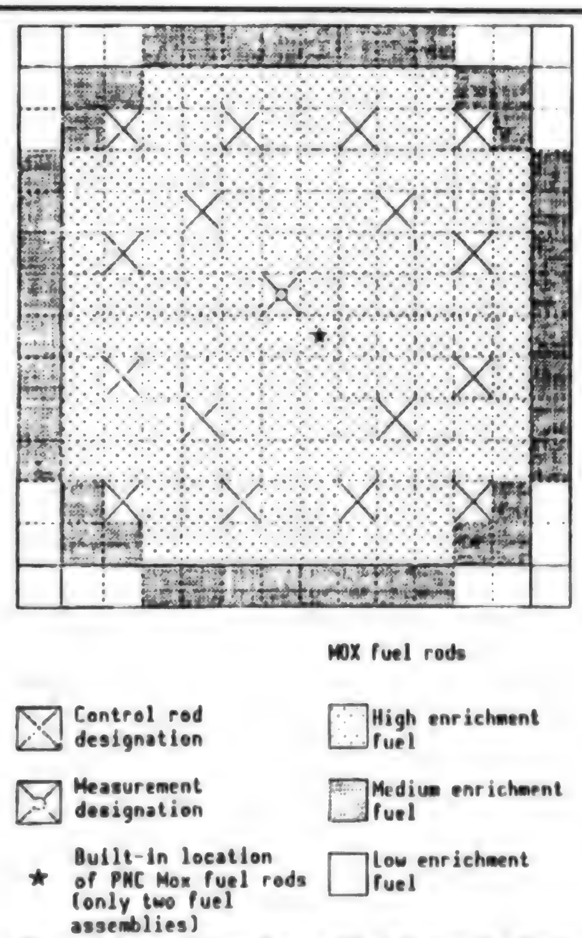


Figure 5. No. 1 Mihama Reactor Mixed Nuclear Fuel Design

[passages omitted]

5. Conclusion

Our research indicates the following possibilities: (1) That we may find that because mixed nuclear fuel contained plutonium from the beginning that the characteristics of plutonium in terms of core behavior are better than uranium, but that by adopting the right fuel assembly and core design that a design almost identical with a uranium core could be achieved with a light water reactor, and (2) that by appropriately weighing the advantages of plutonium content we could achieve a highly reliable mechanical design. These points have been validated both inside and outside Japan based on actual performance. In order to create a stable supply of energy in the future for Japan which is poor in energy resources, it would be advantageous to promote the use of mixed nuclear fuel.

LWR Processing of Spent Fuel

94FE0521D Tokyo GENSHIRYOKU KOGYO in Japanese Jan 94 pp 31-36

[Article by T. Tateuchi, Japan Nuclear Fuel, Ltd]

[Text]

1. Introduction

In the development of nuclear energy, Japan has steadfastly adhered to a nuclear fuel recycling policy in which spent fuel is reprocessed from the initial stages, and the recovered plutonium and uranium are recycled and reused as nuclear fuel.[1]

Establishing a reprocessing technology, which is an essential part of nuclear fuel recycling, is particularly important to a resource poor country such as Japan, with the point being that newly produced plutonium can be used as a purely Japanese energy resource and improve the energy security of the country. In addition, the reprocessing of spent fuel lessens the amount of radioactive waste to be treated and disposed of and makes it easier to manage radioactive wastes. Thus from the environmental point of view, there is no time to lose in perfecting a reprocessing technology.

The development of LWR reprocessing got underway in Japan in September 1977 when the PNC (Power Reactor and Nuclear Fuel Development Corp.) Tokai Reprocessing Plant started separating the first spent fuel, and continued until December 1992, wherein approximately 680 tons of spent uranium fuel, including the fuel of the prototype "Fugen" advanced thermal reactor, was reprocessed. Japan is second only to France among countries reprocessing spent fuel.

In December 1992, the JNF (Japan Nuclear Fuel, Ltd.) Rokkasho Reprocessing Plant was designated to be the first privately owned reprocessing plant in Japan. Construction of the plant began in April 1993. The foundation of the plant is currently being built with a completion date of January 2000. JNF has attached maximum importance and taken on the responsibility of ensuring that construction of the Rokkasho Reprocessing Plant proceeds smoothly including safety during construction.

2. Current Status Surrounding Reprocessing

As far as other commercial reprocessing plants in operation that target LWR fuel, France has two such facilities, the UP2-400 (400t/year uranium processing capacity) and the UP3 (800t/year uranium processing capacity), giving France a total annual uranium reprocessing capability of 1200 tons.

There are two other plants scheduled to go into operation in the future. The first is a facility called the UP2-800 in France, which basically adds two new facilities, a separation-dissolving facility and a first extraction cycle facility, to the UP2 reprocessing facility already in existence to increase its capacity by 400 tons, and a second reprocessing facility known as THORP, which is scheduled to go into operation in England (1200t/year uranium processing capacity). The UP2-800 is currently undergoing cold tests and is scheduled to begin hot tests in 1994. Construction of the THORP was completed in 1992, and uranium tests were begun in September 1993. A second round of public debates started in August 1993 was completed in October, which leads us to believe that hot tests are to begin soon. When these plants are operating, the total annual uranium processing capacity of light water reactor fuel will reach 2,800 tons. The UP2-800 will be used mainly for domestic reprocessing in France, and will have the capability to reprocess 2000 tons of uranium per year from foreign countries such as Japan.

The annual amount of spent uranium fuel that will be generated in Japan by the year 2000 is projected to be 1100 tons, and 1360 tons by the year 2005, which will be considerably more than the reprocessing capability of the UP3 and THORP, so that is why we felt it urgent for Japan to build its own reprocessing system.

3. Tokai Reprocessing Plant[3,4]

(1) Construction of Tokai Reprocessing Plant

People have been crying about the need for reprocessing plants in Japan since the 1950s, but because Japan lacked the experience and technical expertise, it decided to import technology with a proven track record from abroad and build its own facility. In April 1962, a special subcommittee on reprocessing within the Atomic Energy Commission of Japan drafted a report recommending that a plant with a reprocessing capability of 0.7-1.0 ton/day be built around the year 1968.

Heeding this recommendation, the PNC, which was known at the time as the Nuclear Fuel Corporation, developed plans for the construction of a reprocessing plant in Tokai, Ibaragi Prefecture, and started building the plant in 1971 based on a detailed design provided by the French company, Saint Gobain, and completed the

facility in 1974. After completing water channeling tests, chemical tests, and uranium tests, the plant underwent hot tests with JPDR spent fuel. In December 1974, a revision of the nuclear reactor law made it necessary for the plant to submit to an official inspection on "performance" of the reprocessing facility before going into commercial operation. After two publicity campaigns conducted by the PNC, the plant underwent an official inspection from April to December 1980, and in December it passed the inspection and the plant was put into full-scale operation the following year, which it has maintained up to the present time.

(2) Profile of Tokai Reprocessing Plant

In Table 1, we list the main specifications of the Tokai Reprocessing Plant. In this section, we will briefly describe the reprocessing steps.

Table 1. Main Specifications of Tokai Reprocessing Plant

Spent fuel in storage	Maximum 140 tons U
Fuel reprocessed Note 1	BWR, PWR spent fuel
^{235}U initial enrichment	Less than 4%
Burnup	Less than 35,000 MWd/t maximum burnup per assembly. Less than 28,000 MWd/t average burnup per day in processing fuel
Number of days to cool	More than 180
Processing capacity	0.7t-U/d
Processing method	Purex method (mixer-settler)
Product	Plutonium nitrate solution Note 2

Note 1: Also capable of receiving/processing advanced thermal reactor spent fuel.

Note 2: Adjacent to Tokai plant is facility developing technology for MOX conversion.

The spent fuel which is shipped by transport cask is removed from the fuel assemblies at a fuel extraction pool, and is then stored temporarily at a storage pool.

The spent fuel is then conveyed from the storage pool to a mechanical treatment process by a fuel conveyor.

Here, a shearing machine employs what is called the chop and leach method. Horizontally placed fuel assemblies are fed forward 3-5 cm at a time by a feeding device, they are clamped in place using a gagging device and cut with a shearing blade. The sheared fuel is then sent by chute to a basket within a dissolving tank. The end pieces of the assemblies are sheared off and removed with a special cutting device.

The dissolving operation is done in maximum 400 kg batches per operation. The gaseous wastes that are produced during the shearing and dissolving operations are released from the main stack after being scrubbed and passed through a high-efficiency filter. The undissolved fuel cladding material (hereinafter called hulls) that remains is put into special stainless steel drums along with the end pieces of the fuel assemblies and is disposed of as high-level radioactive waste.

The dissolved fuel solution is then passed through a pulse filter because it still contains undissolved materials that would have an adverse effect on the final treatment process. These include undissolved residues of fissionable products, shielding fragments produced during shearing, and undissolved material attached to the surface of the fuel pin (cladding).

After the acid content and uranium content of the solution are adjusted, the fissionable products (hereinafter called FP) are separated as an aqueous solution by a method in which the solvents are extracted using a mixer-settler as the extractor. After the aqueous solution is heat-concentrated in high-level radioactive waste evaporators, it is disposed of as a concentrated high-level radioactive waste.

After the fissionable products are separated, the mixer-settler is used again to separate and purify the uranium and plutonium, respectively, with the uranium solution being made into a uranium trioxide powder after heat concentration and denitration, and the plutonium solution being heat-concentrated and made into a plutonium nitrate solution.

(3) Principal Technical Developments

Since the Tokai Reprocessing Plant started conducting hot tests in 1977, it has had to be shut down a number of times due to problems with hot nitric acid corrosion of large equipment such as the acid recovery evaporators and dissolving tanks that were not originally planned in the design stage, but it has been resolving these problems one at a time by developing new materials and technologies.

The acid recovery evaporators experienced corrosion leakages in the first generation of evaporators in August 1978, and again in the second generation of evaporators in February 1983. In order to try to improve the corrosion resistance of the evaporators, the PNC selected a Ti5Ta alloy and a zirconium material, each having excellent corrosion resistant properties in an environment where very hot and highly concentrated nitric acid is used, and did some mock-up tests to study and evaluate the material, and then developed a joining

technology for dissimilar materials in order to introduce these materials in reprocessing. During the first officially scheduled plant stoppage between June 1989 and September 1990, the acid recovery evaporator was replaced with a new one made of Ti5Ta alloy as part of preventative maintenance.

There were also two incidents that took place in February 1983 and April 1984, respectively, involving the corrosion of two dissolving tanks. These were resolved by remote repairs and installation of a new tank done with much finer welding seams.

In other areas, the PNC developed a way of reducing the amount of radioactivity released into the environment, and another for solidifying low-level radioactive waste and waste solvents. Through this work, the PNC has developed a trained technical staff with valuable experience, and forming a technology base of domestic manufacturers. Some of the findings that came out of these technical developments have found their way via consulting into the design of the Rokkasho Reprocessing Plant.

4. Rokkasho Reprocessing Plant

(1) Basic Specifications and Technologies Adopted

In Table 2, we list the basic specifications of the Rokkasho Reprocessing Plant.

Table 2. Basic Specifications of Rokkasho Reprocessing Plant

Reprocessing method		Reprocessing LWR spent fuel by Purex method
Reprocessing capacity	Maximum annual processing	800 t-U
	Maximum daily processing	4.8t-U
Spent fuel specifications	Average enrichment	Less than 3.5%
	Cooling period required until reprocessing plant acceptance	More than 1 year
	Cooling period required until shearing can begin	More than 4 years
	Maximum burnup	55,000 MWd/t
	Average burnup per day	Less than 45,000 MWd/t

With an annual uranium reprocessing capability of 800 tons, the plant has been designed from the perspective of safety first, followed by reliability, and cost, based on a long-term outlook of nuclear power plant capacity and the amount of spent fuel that will be generated.

With a policy of using the best available technologies either inside or outside Japan, we studied and evaluated both foreign and domestic reprocessing technologies. We were led to importing reprocessing technology from France on account of the fact that it had the best performance and reliability with regards to the main equipment used in reprocessing light water reactor fuel, that it had experience building two advanced reprocessing plants with an 800-ton annual processing capacity, and furthermore, that the technology base was the same as the Tokai Reprocessing Plant, which adopted the chop and leach process as its pretreatment process and the Purex method as its separation process. Other technologies imported from abroad include depressurized evaporation technology from the United Kingdom and scrubbing technology from Germany.

The domestic technologies that have gone into the Rokkasho Reprocessing Plant include technologies developed by the PNC such as the vitrification of high-level radioactive waste and plutonium denitration. Table 3 gives a summary of the technologies used in the Rokkasho plant.

Table 3. Technologies Used in Rokkasho Reprocessing Plant

Main processes (majority)	French technology (SGN)
Main processes (depressurized evaporation)	British technology (BNFL)
Main processes (iodine removal)	German technology (KEWA)
U, Pu denitration	Domestic (PNC, Mitsubishi Material, Toshiba)
Spent fuel pool	Domestic (Hitachi, Toshiba, Mitsubishi Heavy Industry)
High-level waste vitrification	Domestic (PNC, Ishikawajima-Harima Heavy Industries)

Stainless steel is generally used throughout the plant, but more corrosive resistant materials such as zirconium are also used in certain areas depending on use.

The JNF has concluded a basic cooperation agreement on technology with the PNC concerning the construction and operation of the reprocessing facility. The PNC will work with the JNF over a wide range of areas including offering technologies it has developed and experience gained at the Tokai Reprocessing Plant in addition to exchanging technical staff for the purpose of training JNF staff or conducting joint research.

The PNC have made it a policy to defer the engineering design, fabrication, and installation of equipment to domestic companies because of the quality of Japanese engineers and the similarly high level of quality control, and it has also decided to do actual-scale prototypes before equipment fabrication in order to improve the reliability even further.

(2) Profile of Rokkasho Reprocessing Plant Technology and Equipment

In Figure 1, we show a schematic drawing of the main process in the Rokkasho Reprocessing Plant.

(i) Spent Fuel Receiving and Storage Plant

The spent fuel, which has been stored and cooled at a nuclear power plant, is put into casks and shipped to the

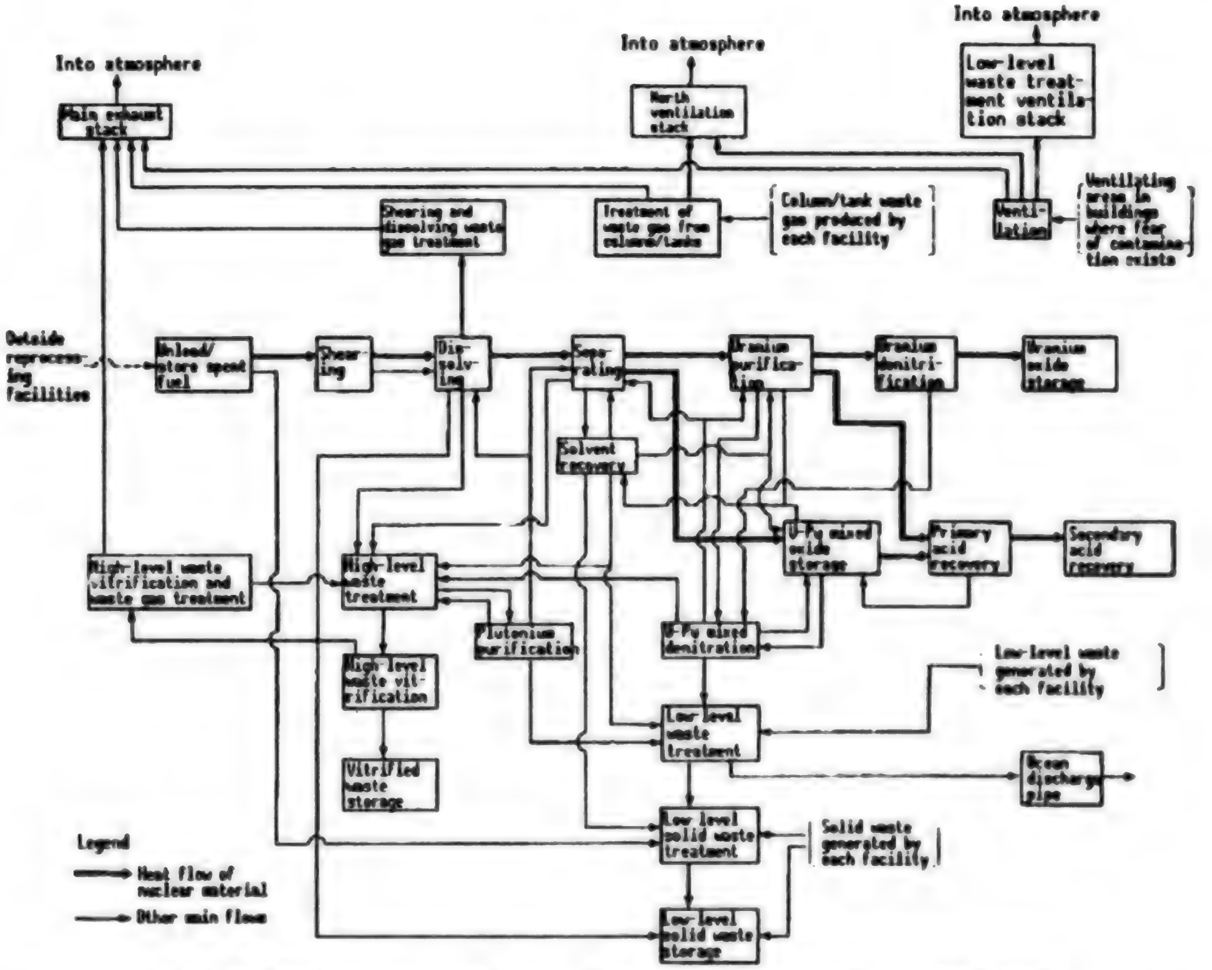


Figure 1. Schematic Diagram of Primary Processes in Rokkasho Reprocessing Plant

reprocessing facility. The fuel assemblies, after being removed from the casks, are delivered to fuel storage pools where they are stored in fuel storage racks. The fuel storage pools are lined with stainless steel in order to prevent leakage.

The fuel storage pool consists of BWR fuel, PWR fuel, and combined BWR and PWR fuel, and has a total storage capacity of 3,000 tons.

After the spent fuel assemblies have sat for the required cooling period, they are sent to a shearing facility in single baskets.

(ii) Shearing Facility

This facility is configured with a fuel supply plant and shearing plant to form two systems.

The spent fuel assemblies are transported by basket conveyors to a fuel supply cell where the assemblies are removed within the cell by a laterally moving crane. The fuel assemblies are continuously fed to a shearing plant where they are chopped into pieces about 3 cm long by a shearing blade.

The sheared fuel fragments are sent to a dissolving tank and the end pieces are sent to an end-piece acid scrubbing tank by a special conveyor pipe.

Nitrogen gas is blown into the shearing machine during the operation in order to prevent the ignition of zirconium powder that is produced by shearing the fuel cladding.

(iii) Dissolving Facility

The shearing facility is configured with dissolving plant and clarifying and weighing plant to form systems.

The sheared fuel fragments go to a dissolving tank where they are dissolved in nitric acid. The dissolving tank used is a continuous-type dissolving tank that was developed in France and used at the UP3 reprocessing plant.

The device consists of the tank itself and a wheel-like device having 12 buckets, and is designed so that fuel is continuously dissolved as the wheel turns, giving it a processing capability greater than the conventional batch method. The sheared fragments from the shearer, which are loaded into buckets, are submerged by the action of the wheel into the nitric acid for a fixed period of time causing the fuel pellets to dissolve and leaving behind only hulls.

When there is a high leftover concentration of spent fuel to be dissolved, a soluble neutron absorbent is added to the tank to dissolve the remaining spent fuel.

The solution in which the spent fuel is dissolved overflows continuously from the dissolving tank and is discharged into an iodine purging tank. After the residual iodine in the dissolving solution is purged by nitrous oxide (hereinafter NO_x) as it is heated in the two-phase iodine purging tank, it is sent to a clarifier.

The dissolving and iodine purging tanks are made with zirconium in order to prevent any corrosion of the materials.

When the hulls that are left behind in the bucket after the fuel is dissolved reach the hull discharge position by the turning of the wheel, the hulls are gravity fed from the dissolving tank into a scrubbing tank and scrubbed in water. After the end pieces have been scrubbed with acid and water, they are put into containers with the hulls and transported to a waste facility for solid waste.

The clarifier is a centrifugal-type device having bowls inside that rotate at approximately 2000 rpm and collect and remove undissolved residues on the inner surface of the bowls. After the undissolved residue collected on the inside surface of the bowl is scrubbed with nitric acid, it is flushed with water and discharged into an undissolved residue recovery tank.

After the acid content of the clarified solution is adjusted at the weighing and regulating tank, it is weighed and sent on to a separating facility.

Weighing and analyzing a sample of the dissolving solution from the weighing and regulating tank is done to verify not only the amount of nuclear material contained in the dissolving solution but also the concentration of ^{235}U and plutonium which is important in terms of criticality control, and to adjust these as necessary.

(iv) Separating Facility

The separating facility is equipped with a plant for separating uranium, plutonium, and FPs, and also a distributing plant for separating uranium and plutonium. It separates spent fuel by a three-step solvent extraction cycle involving extraction, scrubbing, and reverse extraction. The organic solvents which are used include tributyl phosphate (hereinafter TBP) and dodecane oxide, a diluent of that containing about 30 percent TBP.

A pulse column and mixer-settler are used as solvent extractors.

1. The separating plant separates uranium, plutonium, and FPs from the dissolving solution with an organic solvent and removes the FPs.

   After the small quantities of FPs contained in the solution are scrubbed and removed by primary and secondary scrubbing columns, the organic solvent which extracted the uranium and plutonium is sent to a distributing plant.

   After the TBP is removed by the respective TBP scrubbing column and TBP scrubber, the extracted waste containing FPs from the extraction column and auxiliary extractor is sent to a high-level waste treatment plant in the liquid waste disposal facility.

   Pulse columns are used for the extraction column, primary scrubbing column, secondary scrubbing column, and TBP scrubbing column, and mixer-settlers are used for the auxiliary extractor and TBP scrubber. A pulse column has better processing capability and criticality control than a mixer-settler. The pulse columns used at the Rokkasho plant have raised the level of criticality safety along with the neutron absorbent which includes concrete containing boron and cadmium.

2. The distributing plant receives the organic solvent containing uranium and plutonium from the separating plant, separates the uranium from the plutonium, and sends these on to respective purifying facilities. Here, the plutonium is stripped to an aqueous phase using a uranous (quadrivalent uranium) reducing agent to reduce the plutonium to a trivalent uranium.

   After the minute quantity of uranium is removed by the uranium scrubbing column and the TBP is scrubbed with a plutonium solution TBP scrubber, the plutonium nitrate solution is sent to a plutonium purifying facility. After the minute quantity of plutonium is removed by the plutonium scrubber, the organic solvent containing uranium from the plutonium distributing column is stripped of uranium by a uranium extractor using diluted nitric acid. Following this, the TBP is removed with the TBP scrubber, and after being enriched in a uranium enrichment drum, it is sent to the uranium purifying facility.

   Pulse columns are used for the plutonium distributing column and uranium scrubbing tower, and mixer-settler extractors are used for the plutonium scrubber, the uranium stripper, the uranium solution TBP scrubber, and plutonium solution TBP scrubber.

(v) Purification Plant

The uranium puiification plant performs uranium purification (extracting, scrubbing, and stripping) and enrichment work.

The plutonium purification plant performs plutonium reoxidation, purification, (extracting, scrubbing, and stripping) and enrichment work. The valence of plutonium is controlled using NO_x gas as the oxidizing agent and hydroxylamine nitrate (HAN) as the reducing agent.

Pulse columns are used as the extraction column and scrubbing column in plutonium purification, and mixer-settler extractors are used as the extractor and scrubber. The uranium purification plant is not as restricted as the plutonium purification plant in terms of criticality control, so it uses mixer-settler extractors. The plutonium enrichment drums are made out of zirconium.

(vi) Denitration Facility

This facility consists of a denitration plant and U-Pu mixed denitration plant.

1. The uranium denitration plant receives the uranium nitrate solution from the uranium purification plant, and after continuous enrichment in enrichment drums, it is sent to the denitration system.

   The denitration tower of this facility is a piece of equipment that blows a current of air into the tower from below to effect a flow of UO_3 within the tower, and by spraying atomized uranium nitrate into the fluidized bed heated to 300°C, produces UO_3 powder by heat decomposition.

   The UO_3 powder that is produced is overflowed continuously from the denitration tower into storage containers which are filled and sealed. The storage containers are placed in storage baskets and shipped to a uranium oxide storage facility in the product storage center.

2. The U-Pu mixed denitration plant receives the plutonium nitrate and uranium nitrate solutions, and after mixing the two solutions together in a 1:1 ratio, sends the solution to the denitration plant.

   The mixed solution is received at the denitration plant and is supplied to denitration plates. The plates with the solution are irradiated by microwaves from a microwave oscillator which vaporizes, enriches, and r...oves the nitrates from the solution, leaving beh...d a mixed U-Pu oxide powder (PuO_2, UO_3).

   Next, the U-Pu mixed powder is baked in a sintering furnace, and then further reduced in a reducing furnace to form MOX (PuO_2-UO_2) powder. The MOX fuel is placed in powder drums and the powder drums are sealed in storage containers and sent to a U-Pu mixed storage plant. The sintering and reducing furnaces are both made out of a high-temperature resistant nickel-based alloy (hastelloy).

(vii) Acid and Solvent Recovery Facility

This facility consists of a recovery plant for acid produced in the reprocessing facility, and a solvent recovery plant that recovers spent organic solvents produced in the separation and purification facilities.

1. The acid recovery plant consists of a primary and secondary acid recovery system. The spent nitric acid produced by the high-level waste treatment plant is evaporated in the primary recovery system by an evaporator and a fractionating column, and the recovered nitric acid is reused in the purification facility, etc. The spent nitric acid produced by the purification and denitration facilities undergoes similar treatment in the secondary recovery system.

   The evaporator and fractionating column are made out of stainless steel, but operating temperature has been lowered by depressurized evaporation in order to moderate the corrosive environment.

2. The solvent recovery plant consists of solvent recycling and solvent treatment systems. The solvent recycling system uses a solvent scrubber (mixer-settler) to remove decomposed TBP matter scrubbed with sodium carbonate and nitric acid from the spent organic solvent produced by the separation and purification facilities, and the organic solvent is reused again in the separation and purification facilities.

   A portion of spent organic solvent is periodically taken from the solvent recycling system and sent to an evaporator and solvent fractionating column in the solvent treatment system to be evaporated, and the recovered diluent is then reused again in the separation and purification facilities, and the organic solvent is sent on to the solvent recycling process. The evaporator and fractionating column do their work in a depressurized state due to fear of spontaneous combustion of the organic solvent.

(viii) Waste Facility for Radioactive Wastes

With regard to the treatment and disposal of the gas, liquid, and solid waste produced at the Rokkasho Processing Plant, we have tried to keep the dose equivalent lower than that in the general public firstly, by not allowing the general dose equivalent to exceed the dose equivalent set by law as radioactive material is released, and secondly, by trying to keep the release of radioactive substances within reason to the lowest minimum possible, and thirdly, by adopting reliable technologies and incorporating the best of these technologies in each of the processing plants according to the scientific and physical properties of radioactive material.

5. Conclusion

The Rokkasho Reprocessing Plant will be completed and put into operation in the year 2000. The technology and operating experience acquired at the Rokkasho and Tokai plants will be put to use again in the future in a second privately run reprocessing plant.

Bibliography

(1) Atomic Energy Commission of Japan Nuclear Fuel Recycling Subcommittee Report, August 1991.

(2) *Genshiryoku Pokketobukku*, 1993.

(3) 20-Year History of PNC.

(4) *Genshiryoku Pokketobukku*, Ohmu-sha.

Fast Reactor Reprocessing of Spent Fuel

94FE0521E Tokyo GENSHIRYOKU KOGYO in Japanese Jan 94 pp 37-42

[Article by H. Ojima, Power Reactor and Nuclear Fuel Development Corporation (PNC), Nuclear Fuel Recycling Development Division]

[Text]

1. Introduction

Fast breeder reactor (FBR) fuel reprocessing entails the recovery of plutonium (Pu) for recycling purposes, so it is positioned upstream in the FBR cycle.

The reprocessing of FBR fuel is basically done by the Purex method, which is the same used in reprocessing light water reactor (LWR) fuel, but the spent fuel of a fast breeder reactor is distinctly different from that of a light water reactor in the following ways:

(1) FBR spent fuel contains a higher degree of fissionable products because FBR burnup is higher.

(2) Plutonium content is higher.

(3) Structurally quite different in that FBR has wrapper tubes and wrapping wire.

Therefore, in comparing FBR fuel reprocessing with LWR fuel reprocessing, the following problems must be solved as diagrammed in Table 1.

(1) The removal rate of fissionable products must be improved in order to cut down the release of radioactive material into the environment.

A more effective removal method must be developed for removing the large quantity of insoluble fissionable products in the nitric acid fuel dissolving solution (undissolved residues).

(2) More thought needs to go into the fuel dissolving, extracting, and distributing processes to deal with the stricter criticality control in the dissolving tank, etc., and the increasing size of the equipment.

A more effective solvent extraction system must be developed in order to reduce solvent degradation.

(3) A plan is needed for mechanical pretreatment processes such as removing wrapper tubing.

2. Details of Development

The PNC started developing FBR fuel reprocessing technology in 1975. As we can see from Figure 1 illustrating the steps taken to solve the various problems described above, the PNC decided to integrate test models (basic tests, engineering tests, actual-scale tests) and levels of handling radioactivity (cold, uranium, hot) according to the technology to be developed, and set up a development facility within the Tokai Works for that purpose.

The initial period of development included work at the Engineering Design Facility (EDF) developing processing equipment technology such as a rotating saw and cutting technology (generally used technology at that time) for

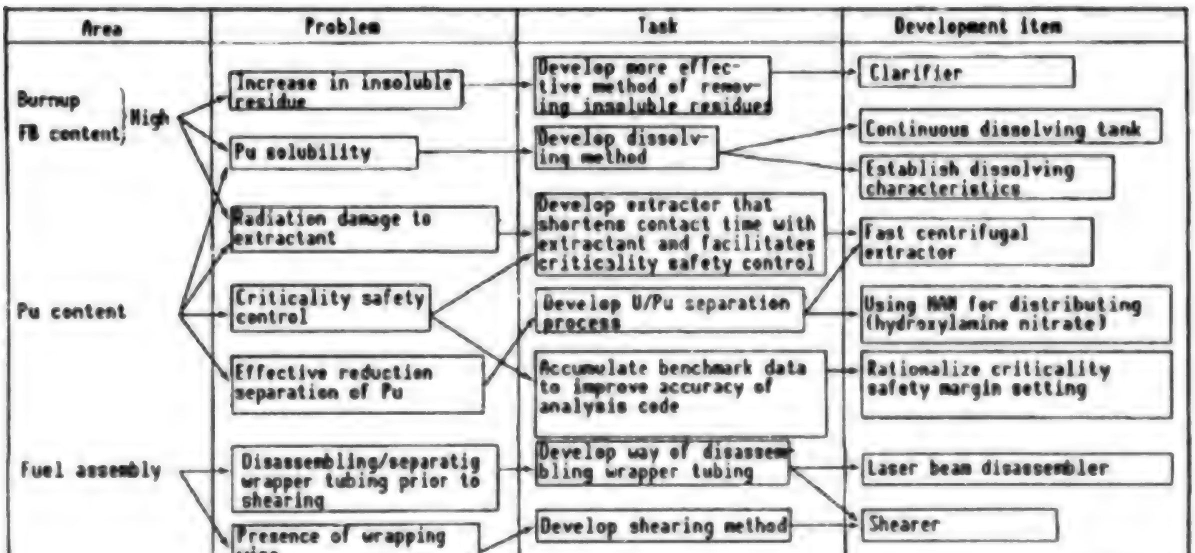


Table 1. Specific Developmental Problems Regarding FBR Reprocessing

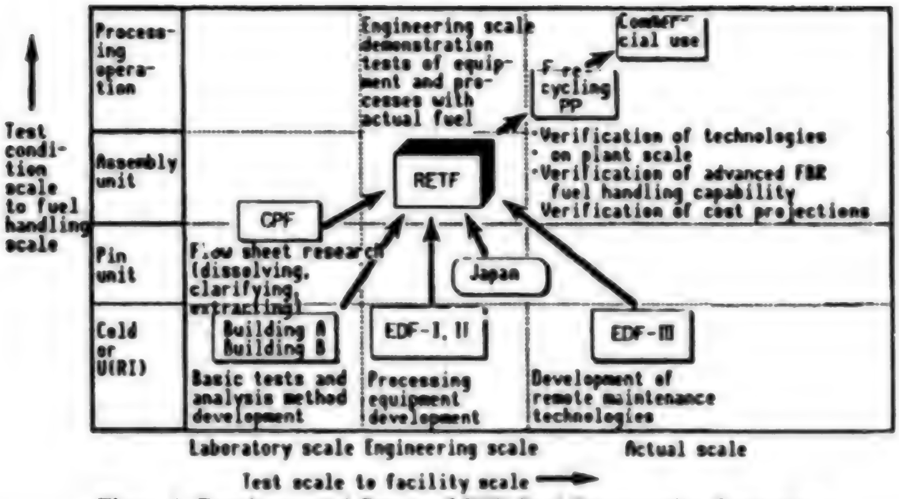


Figure 1. Developmental Stages of FBR Fuel Reprocessing Technology

removing wrapper tubing, a batch-type dissolving tank (technology adopted by Tokai Reprocessing Plant) for dissolving fuel, and a pulse column (generally used technology at that time) for extracting solvent. Since 1982, the PNC started hot lab-scale process foundation tests at the Chemical Processing Facility (CHF) using actual spent fuel from the demonstration "Joyo" FBR.

By the mid 1980s, these technical developments had enabled the PNC to see that the Purex method was applicable to FBR reprocessing.

Later, the PNC decided to raise the level of reprocessing technology even higher, and began an active campaign of incorporating advanced technologies from general industry, and, in accelerating that process, started conducting joint research in 1987 with the Oakridge National Research Laboratory in the United States.

The PNC targeted the following processing equipment to be developed based on technical developments both inside and outside Japan, and built a full scale test facility where it is currently conducting cold mock-up tests.

- Laser disassembler for removing wrapper tubing (enables non-contact cutting and eliminates need of replacing cutting teeth)
- Continuous-phase dissolving tank for dissolving fuel (having large processing capability in view of strict dimension requirements due to criticality control)
- Centrifugal extractor for extracting solvents (being compact and having high processing capability, and reducing solvent degradation due to radiation)
- Large-scale remote cell system for maintenance (various types of equipment installed in one cell having remote maintenance capability; expected to dramatically reduce maintenance time)

In the meantime, processing technologies are also being developed. Hot basic tests are being done on the development of a salt-free technology and a method of extracting both uranium and plutonium in order to pave the way for a more advanced Purex method from the view of trying to reduce radioactive waste and improving nuclear non-proliferation.

In order to confirm the technical feasibility of the above-mentioned new equipment, remote maintenance, and processing technologies, the PNC is planning to build a new testing facility called the Recycle Equipment Test Facility (RETF) that will be used as a place to conduct engineering-scale hot tests with actual spent fuel.

3. Developmental Status

(1) Developing Processing Equipment

(i) Laser Disassembler

The reprocessing of fast reactor fuel starts first with the breaking up and removal of the stainless steel wrapper tubing.

In the initial period of development, the PNC tested a mechanical cutting method, but as development moved forward lasers came into favor and a laser beam disassembler was finally installed at the second EDF building where tests were done on cutting properties and disassembling procedures. At the stage where the applicability of a technology is determined, the PNC conducted joint research with the Oakridge Lab to try to make the device more compact and powerful by relocating the focusing unit, etc., and to collect necessary data.

A detailed design is now being done on that device for the RETF based on those results.

(ii) Shearer

In shearing fuel pin bundles, it is important that there be no constraints and that a way be found for dealing with the wrapping wire, but this process is basically resolved in reprocessing LWR fuel by cutting the bundles by hand.

(iii) Continuous Dissol[illegible] Tank

Sheared fragments (approx 3 cm) come into contact with nitric acid and the fuel portion dissolves to form uranium, plutonium, and an FP nitrate solution.

The continuous dissolving tank can process more fuel than the batch-type method even with the stricter criticality control, but something still has to be done about whether it should have a mechanical drive unit. After studying various possibilities, the PNC decided to adopt the idea shown in Figure 2 which was proposed in joint research with the Oakridge Lab.

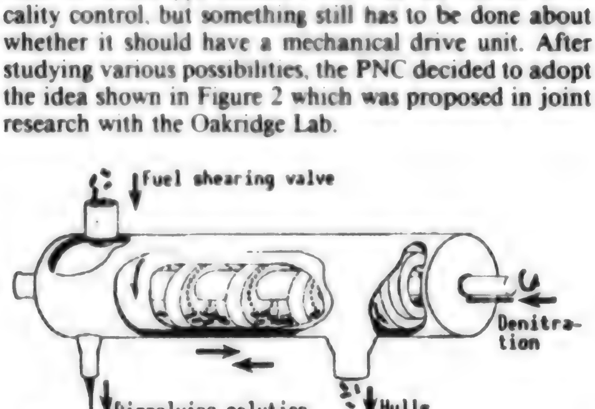


Figure 2. Cylindrical Rotating-Type Dissolving Tank

The tank has a double-cylinder exterior in order to contain the radioactive material.

The interior of the tank, called the drum, is divided into eight stages by helical separating plates. The drum is supplied from one end with sheared fuel fragments, and from the other with heated nitric oxide. The sheared fuel fragments proceed to each successive stage by the drum being shaken regularly to bring the sheared fuel fragments into better contact with the nitric acid, and also by the regular rotation of the drum.

In order to evaluate the behavior of the fragments and particles within the drum, we conducted tests with a plastic model to verify that the fragments were advancing smoothly and made criticality calculations while the FBR fuel was being dissolved. Using these results, we installed an RETF prototype dissolving tank at the Oakridge lab and used that to verify remote maintenance characteristics and to conduct tests with uranium.

(iv) Centrifugal Clarifier

The stable operation of the solvent extraction process is contingent upon protecting the operation from minute insoluble residue consisting mainly of platinum group fissionable products left behind in the fuel dissolving solution. Particularly in the case of high burnup spent fuel from reactors like the FBR, a more highly efficient removal method than that used in LWR reprocessing has to be found because the amounts of insoluble residue are much higher.

Using a test clarifier installed at the No. 2 Applied Testing Facility, we conducted a simulation test using aluminum as the undissolved residue and verified that the machine could effectively remove minute particles smaller than a micron in size, and also that maintenance could be done remotely (Figure 3).

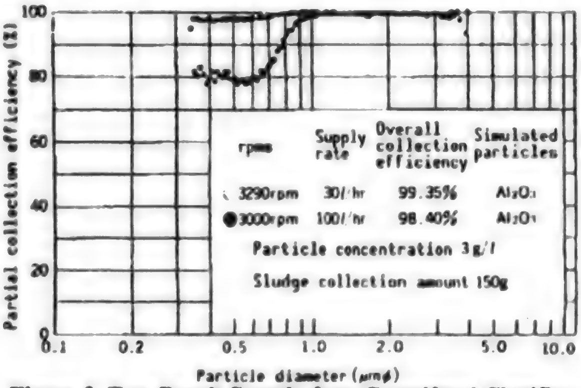


Figure 3. Test Result Sample from Centrifugal Clarifier

(v) Centrifugal Extractor

For separating uranium, plutonium, and fissionable products, tributyl phosphate (TBP) is used as an extractant.

Fuel solubility and contact time with the solvent in a centrifugal extractor is far shorter than conventional extractors such as mixer-settlers or pulse columns, which means that even though it is more compact, the centrifugal extractor achieves adequate processing and produces far less solvent degradation. In addition to compactness, the fact that it takes much less time to start and stop has a major impact on reducing cost.

On the other hand, because the structure of the machine itself is more complex than other extractors, the PNC was concerned with maintainability and began to conduct uranium tests in 1986 at the No. 2 Applied Testing Facility. Additional uranium testing was done in joint research with the Oakridge Lab based on a partial mock-up of the RETF solvent extraction process.

The extraction of uranium by a centrifugal extractor system agreed well with the calculated values in an analysis code developed by the PNC called the "MIXSET" (Figure 4).

(vi) Electrolyzer

As part of the work to develop a salt-free process aimed at lowering the amount of waste containing salt, the PNC has made progress developing and perfecting the technologies for solvent cleansing electrolytes, i.e. hydrazine carbonate.

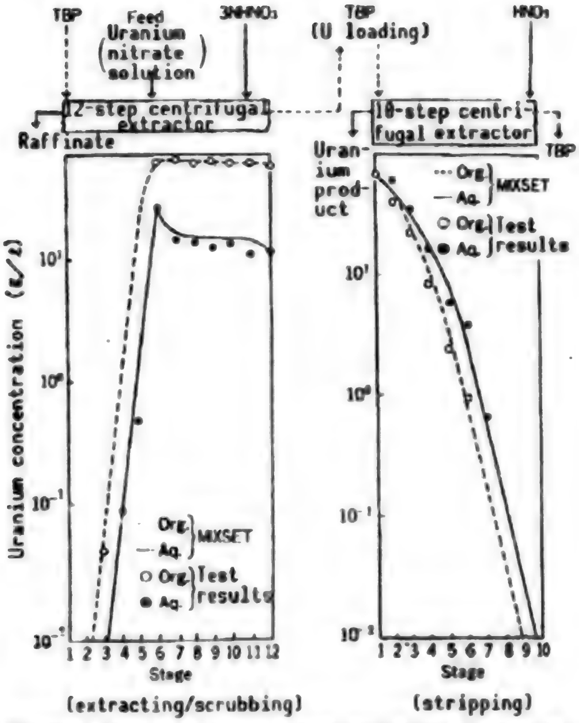


Figure 4. Uranium Concentration Profile With Extracting/Scrubbing by Centrifugal Extractor and With Stripping

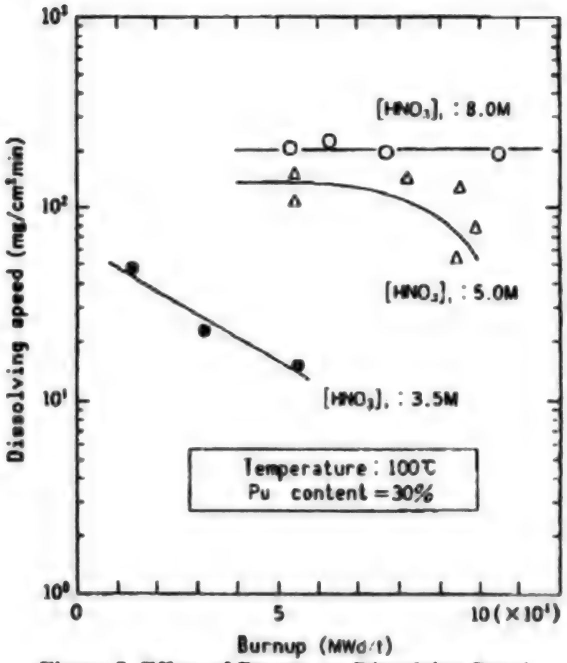


Figure 5. Effect of Burnup on Dissolving Speed

(2) CPF Results

The hot process bench tests conducted at the CPF have helped to confirm that FBR fuel can be reprocessed by the Purex method. The main results obtained so far are listed below:

1) Dissolving Characteristics of Fuel—Using various types of fuel up to a maximum burnup of 94,000 MWd/t, we ascertained, as can be seen in Figure 5, the effect of nitric acid concentration, dissolving temperature, and burnup has on dissolving speed.
2) Insoluble Residue in Dissolving Solution—Here, we ascertained quantities, particulate distribution, and composition, etc.
3) Extraction Process—Here, we collected data on the distribution of U/Pu when U/Pu are extracted together in a solution having a high Pu content, and also when hydroxylamine nitrate is used as a Pu reducing agent, and confirmed optimum flow sheet conditions.
4) In combination with extraction tests, we also conducted tests to remove degraded products in spent solvent.

These results have provided valuable information for determining the design conditions of processing equipment described above.

(3) Basic Technologies

In addition to processing and equipment development, the development of basic technologies shared with LWR that are essential to the safe operation of the facility and reducing radiation exposure of workers are examined.

(i) Development of Materials

In order to solve the problem of corrosion in equipment used in reprocessing, broad-based research is now being done on materials and how they are structurally incorporated in the environment and equipment in which they are used. Using the results from using Ti5Ta in the acid recovery evaporators at the Tokai Reprocessing Plant, the material corrosion problem has been nearly eliminated.

(ii) Development of Remote Maintenance Technology

In order to reduce worker exposure and improve the operation rate of the plant, the PNC has been working on the technologies needed for a large-scale remote maintenance cell like that shown in Figure 6. Using both the findings obtained in joint research with the Oakridge Lab, and technology developed by the PNC on a dual-arm manipulator and rack system, we are laying the foundation for a new remote maintenance technology.

(iii) Analysis Technology

The PNC has already perfected a technology for analyzing the FBR fuel reprocessing tests in the RETF. That work is currently focusing on trying to make in-line analysis and other types of analysis less labor intensive and faster while also paying attention to the needs of the process.

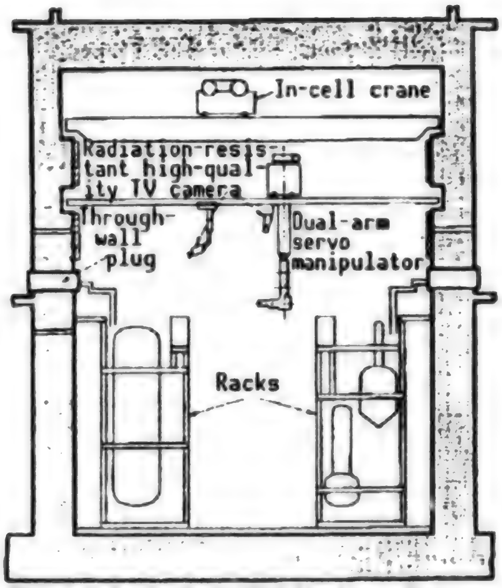


Figure 6. Large-Scale Cell Used as a Remote Maintenance System

5. Recycle Equipment Test Facility (RETF)

The RETF is being built with the following goals in mind: to integrate the results described on technological developments, to perform engineering-scale hot tests on technologies relating to processing and new equipment with actual spent fuel assemblies from the prototype "Monju" FBR and other reactors, and to collect engineering data for demonstrating the validity of technologies and for future facilities. In order that it may have the capability to perform long-term R&D on reprocessing technologies, the RETF is being designed with the ability to deal flexibly with process changes and to perform effective analysis and measurement. Figure 7 is a bird's-eye view of a large-scale test cell in the RETF.

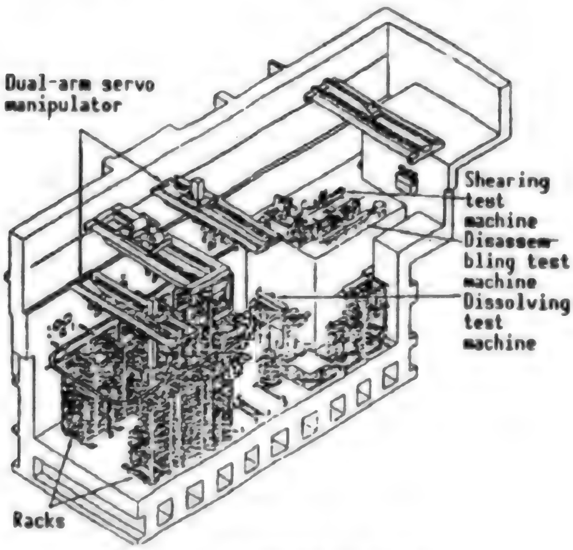


Figure 7. RETF Test Cell

The RETF is scheduled to be built adjacent to the Tokai Reprocessing Plant. In August 1993, it received authorization to make modifications in the reprocessing facility equipment. The construction plans for the future are moving forward with a goal of commencing hot tests by the year 2000.

6. Plan for Future Development of Technologies

Below, we list the three basic developmental goals regarding FBR reprocessing in the future:

(a) To make FBR reprocessing more economical than current LWR cycle (improve cost)

(b) To gain public understanding regarding safety and waste problems (improve safety and environmental protection)

(c) To erase fears surrounding the handling of large quantities of Pu (increase nuclear proliferation resistance)

In Figure 8, we show the main technologies that must be developed in order to achieve these developmental aims. Below, we list the developmental items that are expected to have the biggest impact on achieving these goals:

(1) Process optimization by developing common extraction method for U/Pu/Np and by lowering product DF

Contributes to developmental goals (a), (b), and (c)

(2) Salt-free technology and complete solvent recycling technology (In Figure 9, we show conceptually how a salt-free technology will reduce waste)

Contributes to developmental goals (a) and (b)

(3) TRU separation and recovery technology

Contributes to developmental goals (b) and (c)

(4) Automated operation incorporating artificial intelligence

Contributes to developmental goal (a)

(5) Remote maintenance technology optimization

Contributes to developmental goal (a)

(6) More advanced safeguards

Contributes to developmental goals (a) and (c)

(7) Adoption of earthquake-proofing method

Contributes to developmental goals (a) and (b)

The PNC is undertaking these developments in conjunction with the plan to build the RETF, and the findings made therein will be put back into the design of future FBR fuel reprocessing facilities.

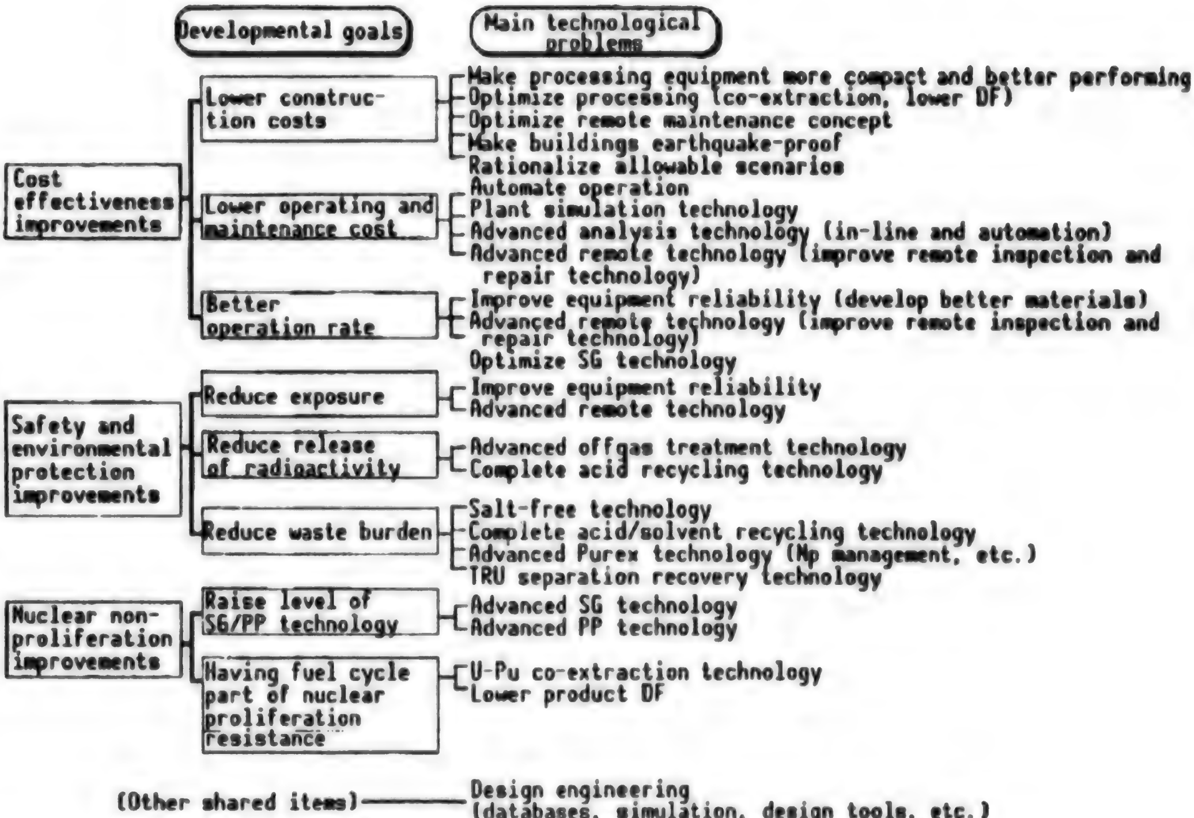


Figure 8. Developmental Goals and Main Technological Problems

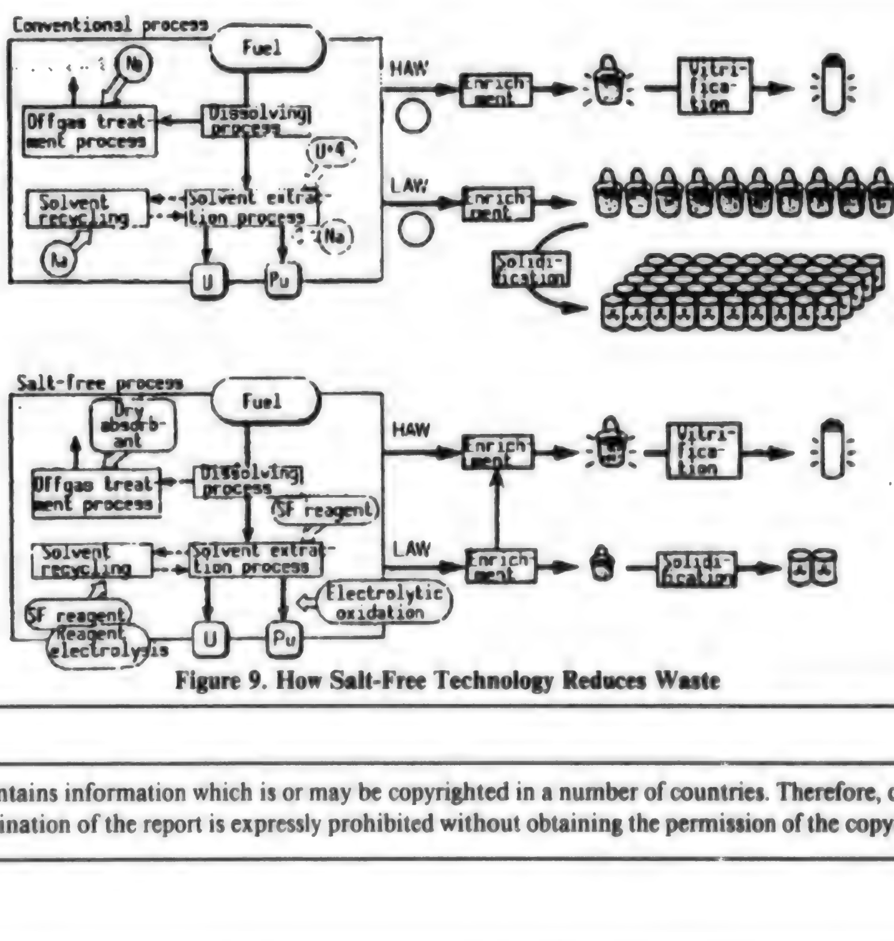


Figure 9. How Salt-Free Technology Reduces Waste

MOX Fuel Processing Technology

94FE0521F Tokyo GENSHIRYOKU KOGYO in Japanese Jan 94 pp 43-48

[Article by H. Oshima, Power Reactor and Nuclear Fuel Development Corporation (PNC), Plutonium Fuel Plant]

[Text]

1. Introduction

In order for the plutonium recovered by reprocessing plants to be reused by a fast breeder reactor (FBR), advanced thermal reactor (ATR), or light water reactor (LWR), it must be first processed into U-Pu mixed oxide fuel (MOX fuel) at a plutonium fuel facility. MOX fuel is basically processed similarly to uranium fuel by a method that regulates the oxide powder and sintering the powder in a high-temperature electric furnace after being press-molded.

The main points of difference between that and uranium fuel processing are that MOX fuel processing requires special plutonium handling technologies with regard to criticality, shielding, exposure, containment, and heating in its processing, that special safeguards and weight controls are needed because it involves plutonium handling, and that there be a uniform mixture of uranium and plutonium in order to ensure a sound state within the reactor while the fuel is being irradiated.

At the Tokai Works of the Power Reactor and Nuclear Fuel Development Corp. (PNC), the No. 1 Plutonium Fuel Development Lab has been trying to develop an MOX fuel and a way of manufacturing that fuel since 1965. The No. 2 Plutonium Fuel Development Lab has similarly been working on MOX fuel manufacturing since 1972 when it first went into operation to manufacture fuel for the "DCA," the prototype "Fugen" ATR, and the experimental "Joyo" FBR. The No. 3 Plutonium Fuel Development Lab has been trying to develop a remote robotic technology for manufacturing MOX fuel in large quantities since 1988 when it went into operation to manufacture fuel for the "Joyo" FBR and prototype "Monju" FBR.

In Figure 1, we show the flow of development of these technologies. In Figures 2 and 3, we show the history of MOX powder conversion and MOX fuel manufacturing, respectively.

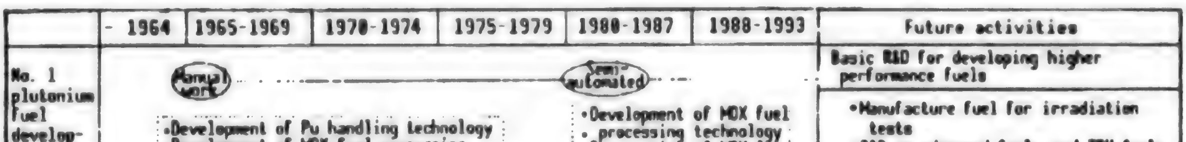


Figure 1. Development of MOX Fuel Technologies

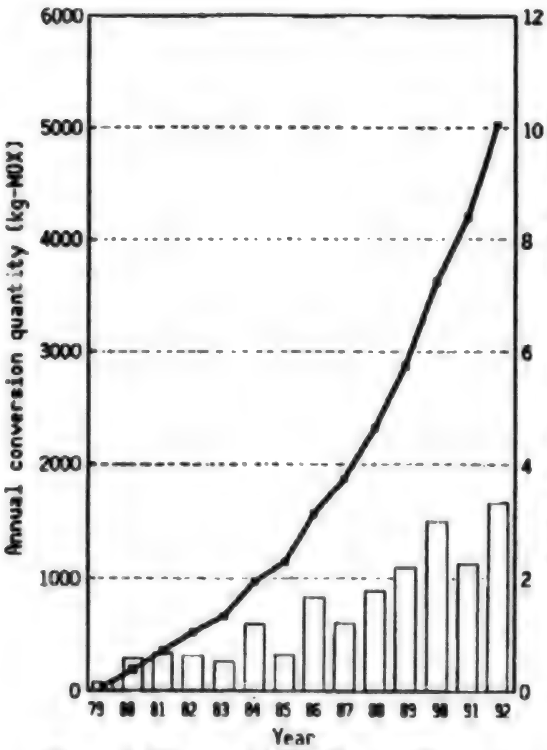


Figure 2. History of MOX Powder Conversion

2. MOX Fuel Design Technology

The fuel of a fast breeder reactor is conceptually quite different from that of a light water reactor. For one thing, it has to be salt-cooled due to the higher linear output during irradiation, and secondly, it has to have a larger final burnup reaching nearly 100,000 MWd/t. In Figure 4, we show a fuel assembly used in the "Monju" FBR. Here, we would like to discuss the technology for designing fuel from the perspective of fuel manufacture.

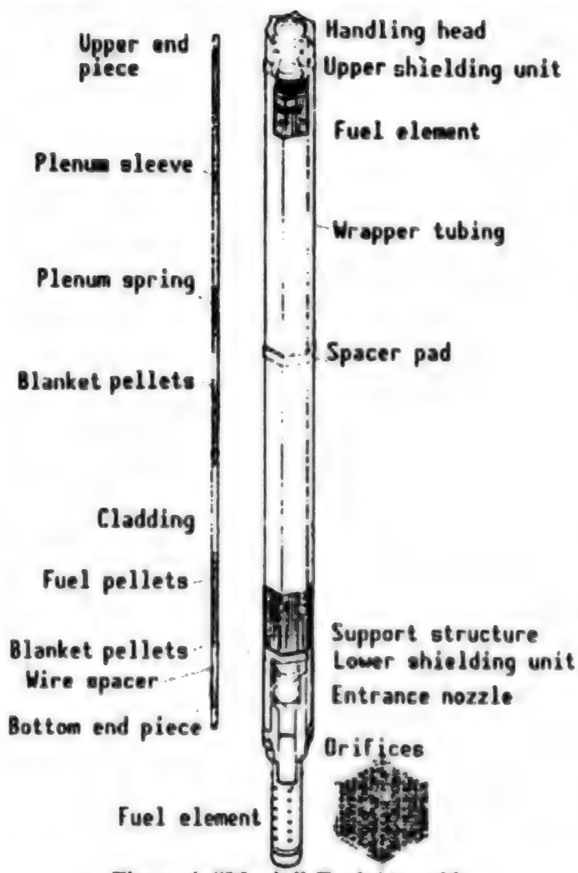


Figure 4. "Monju" Fuel Assembly

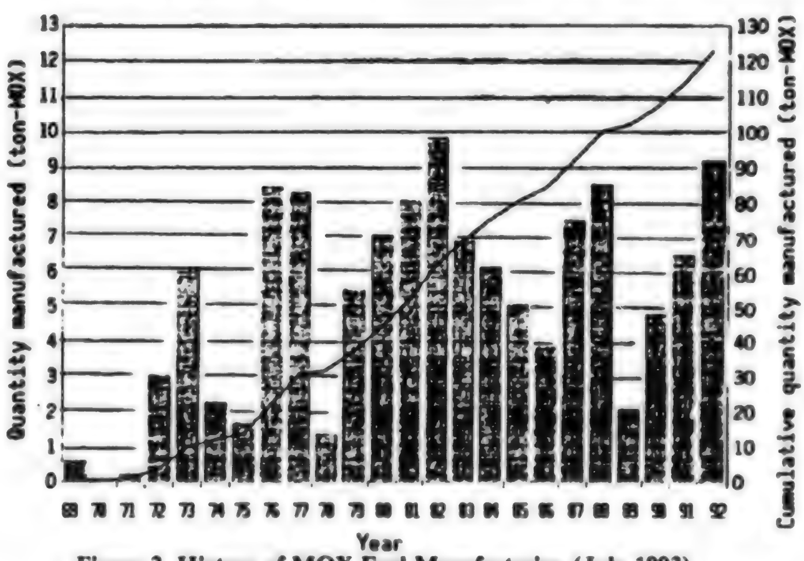


Figure 3. History of MOX Fuel Manufacturing (July 1993)

Plutonium has a range of isotopes extending from Pu-238 to Pu-242, and the isotopic composition in spent fuel will vary according to differences in burnup. The isotope Pu-241, for example, which has high reactivity in the reactor, transforms to Am-241 which is a daughter nuclide with a half-life of about 14 years. There is a major difference in the coefficient factor by which each isotope contributes to reactivity within the core. The differences between each isotope are particularly pronounced in fast breeder reactors, so to gain better control of core reactivity the PNC has adopted the equivalent fussile method in the design and manufacture of fuel, paying particular attention to isotopic composition of plutonium and Am content during core loading. In Table 1, we show the equivalent fussile coefficient for each nuclide

Table 1. Equivalent Fussile Coefficients of Various Nuclide

	LWR	FBR
U-235	+0.8	+0.77
U-238	0.0	0.0
Pu-238	-1.0	+0.44
Pu-239	+1.0	+1.0
Pu-240	-0.4	+0.14
Pu-241	+1.3	+1.50
Pu-242	-1.4	+0.037
Am-241	-2.2	-.0.33

In designing MOX fuel, it is important when mixing uranium and plutonium that there be a large enough margin to allow for fluctuations in Pu content based on the physical properties of the ternary system of Pu-U-O. The evaluation of behavior within the core is also done based on the so-called plutonium spot, whose allowable size during fuel manufacture can be as large as 200 μm in light water and advanced thermal reactors, but is sometimes only about 100 μm in fast breeder reactors.

The burnup in fast breeder reactors is three times higher than it is in light water reactors. That makes it necessary to lower smear density in order to lessen the mechanical interaction between the fuel pellets and cladding which is a cause of pellet swelling in the core during combustion. Doing that requires either a low density pellet or air-borne pellet. The PNC has opted for a low density pellet for the prototype "Monju" FBR.

In FBR fuel processing in particular, the fuel pellets contain a high degree of plutonium, so worker exposure to radiation caused by release of plutonium or daughter nuclide must be contained, therefore, it is necessary that fuel processing be designed with automated equipment so that the fuel assembly, pellet manufacturing, processing, assembling, and inspecting can be done remotely and automatedly.

3. MOX Fuel Processing Steps

(1) MOX Powder Conversion

The plutonium nitrate solution recovered by a reprocessing plant is first converted to oxide powder so that it can be processed into MOX fuel. The conversion of plutonium into oxide powder is done by one of either two processes, namely, the mixed conversion method in which plutonium is made into MOX powder together with uranium, and the single conversion method in which plutonium is made into plutonium oxide powder. The PNC has adopted a mixed conversion method (microwave heating direct denitration method) with a nearly 1:1 plutonium/uranium ratio. This process consists of the following steps: mixing the solution, microwave denitration of the mixed solution, and sintering/reduction. In Figure 5, we show the steps in the MOX powder conversion process.

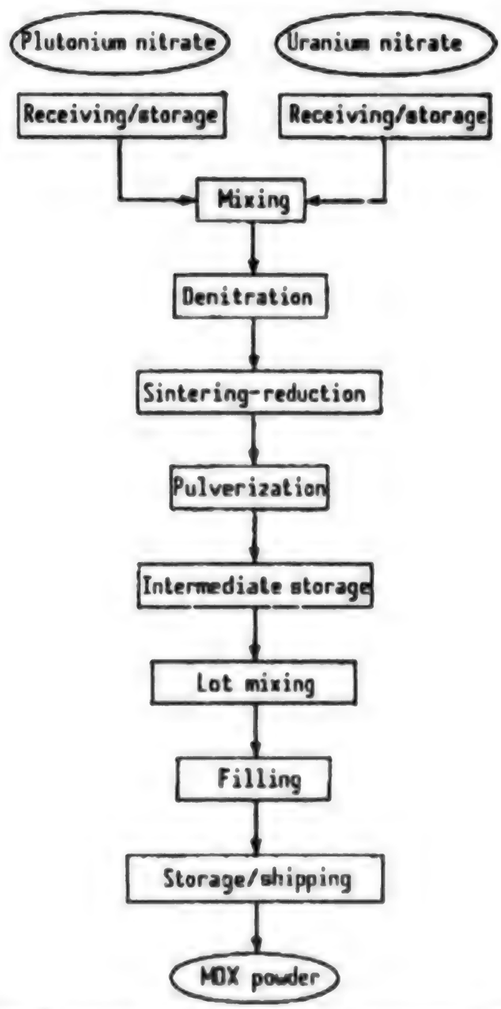


Figure 5. Conversion Process Using Microwave-Heated Direct Denitration Method

The MOX conversion process has some of the same basic features as the pellet manufacturing process described below.

(2) MOX Pellet Manufacturing

In MOX pellet manufacturing, raw powder (MOX powder, or plutonium oxide powder and uranium oxide powder) is received and made into fuel pellets by weighing and mixing this powder with powder recovered by reprocessing, and then pelletizing, shaping, sintering, grinding, and inspecting the powder. The powder, pellets and other materials that do not meet specifications are recovered and reused again as raw powder.

In Figure 6, we show the stages in the manufacturing process of MOX pellets.

The main points of difference between MOX pellet manufacture and uranium fuel pellet manufacture are that (1) the former involves the handling of plutonium so there are restrictions due to concern over criticality safety on the amount handled per unit, (2) the dimensions and layout of the manufacturing equipment are restricted in order to ensure containment of plutonium

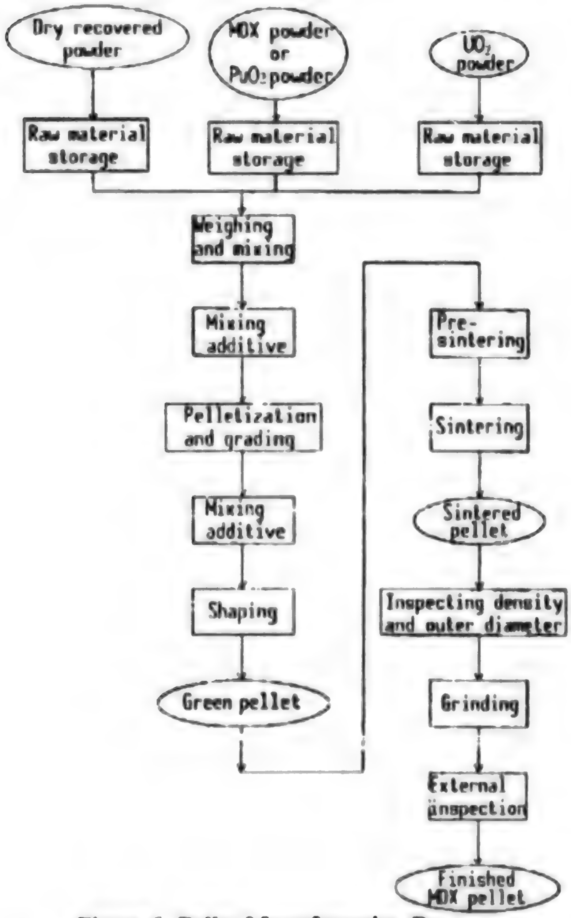


Figure 6. Pellet Manufacturing Process

and enable operation maintenance, (3) the manufacturing equipment has to be automated to limit worker exposure, (4) the physical properties of the powder have to regulated and controlled in order to ensure a uniform mixture of uranium and plutonium, and (5) measures have to be taken to deal with the decay heat from plutonium, etc.

It is important in processing from the regulating powder stage to the shaping stage to have manufacturing control, including a mixing method that achieves the required uniform ratio of uranium and plutonium, and meets specifications regarding the plutonium spot, a pelletization method for stabilizing pellet shape, and a powder control method for minimizing the effect of decay heat.

It is also essential in processing up to sintering stage to have operational control including controlling pellet density, dimensions, and impurity levels. A dry recovery technology is also needed for scrub pellets, which is important from the perspective of making effective use of plutonium.

In manufacturing the initial loading fuel for the "Monju" FBR, the PNC developed the technologies for consistently manufacturing large quantities of pellets with 85 percent lower density while controlling impurities such as nitrogen and carbon.

(3) Fuel Elements and Assemblies

The processing of fuel elements and making of fuel assemblies includes filling the cladding with pellets, welding the end pieces, and making the assemblies. With FBR fuel, there is a wire wrapping process added after the end pieces are welded. In Figure 7, we show the various stages in processing fuel elements and making the assemblies.

The processing up to the welding end-piece stage is done inside a glove box, so the technical requirements are nearly the same as pellet manufacturing. It is particularly important when filling the pellets that the process be managed so as to lessen plutonium contamination at the end of cladding.

In the processing done after the end pieces are welded, the plutonium is sealed inside the cladding, so steps have to be taken regarding criticality safety and worker exposure that would not be taken in uranium fuel processing.

(4) Inspection

All of the processes including the conversion of MOX powder, the manufacture of pellets, the fuel element processing, and the making of assemblies are inspected for quality control, weight control of nuclear material, and process control.

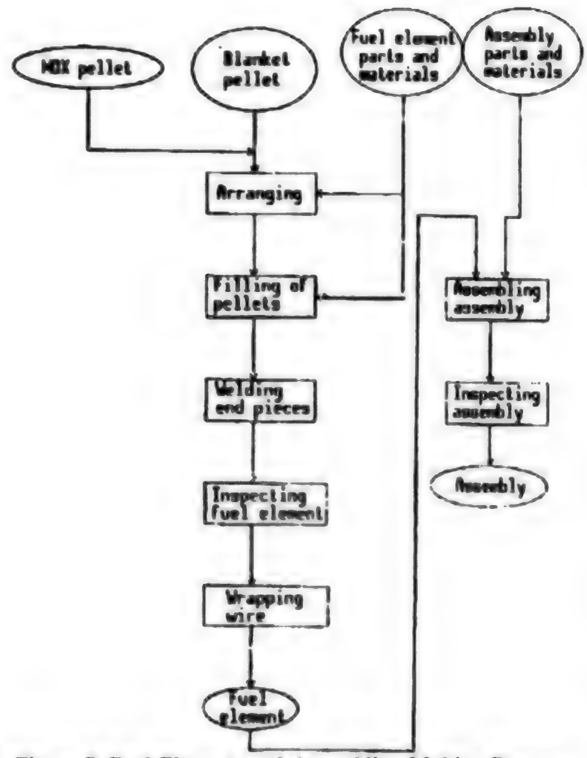


Figure 7. Fuel Element and Assembling Making Processes

Inspections at plutonium fuel facilities are different in that they take place more often due to examining many more details concerning safety, more accuracy is required because plutonium is involved, and most of the inspection equipment is located inside and operated from a glove box.

4. MOX Fuel Processing Facility

Using the fuel manufacturing and plutonium handling experience gained at the No. 1 and No. 2 Plutonium Fuel Development Labs, the PNC has installed and put into operation a plutonium conversion research facility and FBR line at the No. 3 Plutonium Fuel Development Lab. These facilities have a design processing capability of 10 kg Pu+U/day (approximately 1 ton per year) and 5t MOX, respectively.

Here, we will discuss the main technologies used in the No. 3 development lab FBR line.

(1) Pellet Manufacturing Process

This process is limited for safety reasons to a basic quantity of 40 kg MOX and is provided with an intermediate product storage area at the center which is connected on each side by conveyor tunnel with a comb-shaped glove box containing manufacturing equipment. Positioned between the glove box and the storage area is a shielded wall to reduce the dose equivalent amounts in the processing rooms (Figure 8).

With manufacturing equipment, the PNC has tried not only to automate the operation of each piece of equipment, but also to provide remote monitoring-manipulation based on the ITV system as much as possible. The equipment has also been designed to prevent extremely fine dust from being produced when handling powder.

From the safety point of view, we have tried to adopt dry equipment in which moisture is not used in pelletizing powder or grinding the pellets, and have a dry pellet manufacturing process.

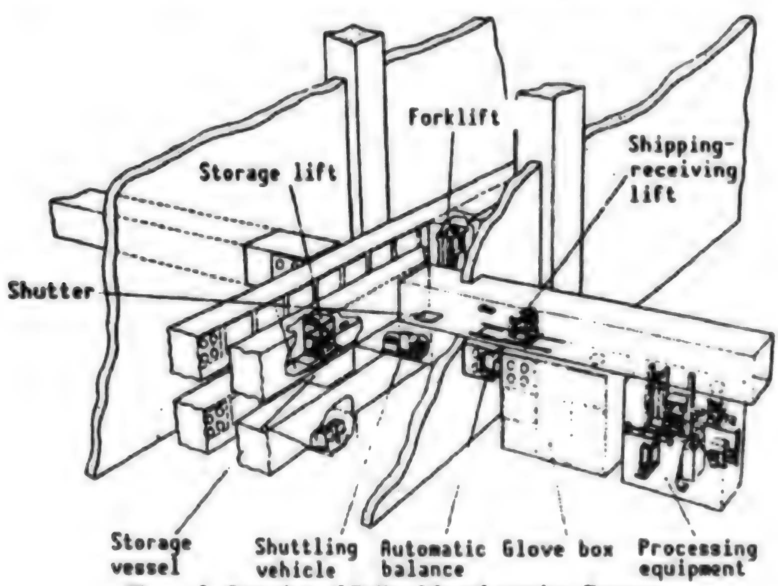


Figure 8. Overview of Pellet Manufacturing Process

Normal operation can also be controlled remotely from the process control room.

(2) Processing and Assembly Process

The equipment used in this process includes filling equipment for MOX pellets, automatic welding equipment for end pieces, contamination inspection equipment, wire wrapping equipment, and fuel assembly equipment. In order to ensure containment of plutonium and to reduce worker exposure, the PNC has sought to automate the operation of each piece of equipment. The equipment has been designed so as to achieve a rational physical flow through the use of fuel element and assembly inspecting and storage equipment, conveyors, and automated cranes.

(3) Operation Management System

The No. 3 Plutonium Fuel Development Lab is managed and operated by a three-tiered computer system. The first tier, the central management system, oversees the entire facility and has the following functions—criticality control, weight control, production control, and inspection records control.

The second tier, the process control system, is in charge of managing and directing the operation of processing equipment used in each stage of pellet manufacture, processing/assembling, and inspecting, and managing the operating data thereof. The third tier, the equipment control system, is in charge of the operation control of the equipment, and collecting and transferring operating data.

These three systems are effectively linked together to oversee the operation and safety of the facility.

(4) Safeguards

It is essential at the No. 3 development lab in terms of operation of the facility that it have a safeguard concept wherein the automated and remote operations are in conformity with the U.S.-Japan agreement, and that it have a safeguard system that satisfies inspection standards of the IAEA. To achieve this, the PNC has been conducting a lot of joint research with the DOE ever since the facility design stage and has come up with a safeguard system consisting of the following five subsystems:

(1) Weight control system for entire facility within central control system

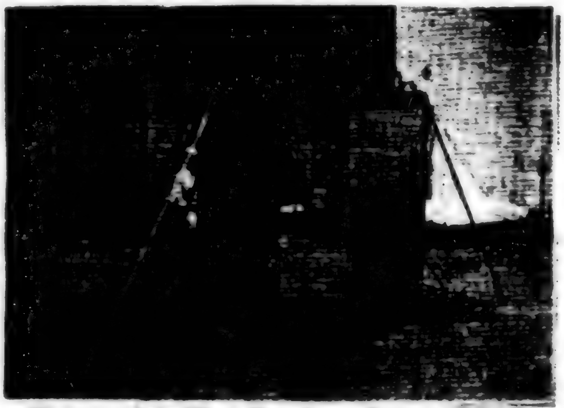
(2) C/S system for storage equipment

(3) Remote NDA system for storage equipment

(4) Remote NDA system within processes

(5) Weight verification system for inspections based on NRTA method

The glove box assay system (GBAS) shown in Photograph 1 is one of the seven NDA measurement systems now in use commercially.



Photograph 1. Glove Box Assay System (GBAS)

5. Future R&D

(1) Manufacturing Technologies

The design, construction, and operation of the No. 3 development lab FBR line has given the PNC a complete MOX fuel manufacturing technology through to commercial application, but problems still need to be resolved including improving safety, reducing exposure, lowering manufacturing cost, and perfecting safeguards when using large quantities of plutonium.

The pellet manufacturing process, for example, needs to manufacture more pellets per unit by either increasing the handling quantity of the criticality control unit or by continuous operation of the processing equipment, have lower exposure risks and reduce labor costs by more automation and remote operations, better reliability in equipment, rationalization of pellet specifications, and measures to prevent dust from being produced.

The processing of fuel elements and assembling fuel assemblies needs a technology for processing new materials that can be used with high burnups, and an automation technology for lowering exposure risks.

The inspection process needs to improve the speed and accuracy of analysis, and rationalize the pellet, fuel element, and assembly inspections.

The PNC believes that it can develop the technologies to resolve these issues in a planned and systematic manner.

(2) Development of TRU Fuel

When we consider plutonium use on a commercial basis, we have to also think about restricting the handling of plutonium in its pure form as much as possible from the view of nuclear proliferation. Furthermore, when we consider the disposal of waste in the future, we also have to think about a way of cutting down the nuclide in the

reactor with a long half-life. In order to solve these problems, the PNC must develop a technology for manufacturing fuel containing TRU. The manufacture of fuel containing TRU requires the same shielding as a reprocessing facility, and all processing should be done remotely from the powder conversion stage to inspecting fuel assembly stage.

While the PNC considers plutonium use on a commercial basis, it is going ahead with work on a TRU fuel manufacture in order to develop new fuel manufacturing technologies.

6. Conclusion

Over a span of almost 30 years going back to 1965, the Tokai Works of the Power Reactor and Fuel Development Corporation has processed more than 120 tons of MOX fuel including various test fuels and FBR and ATR fuels. Moreover, since 1983, the conversion of plutonium has also produced 10 tons of MOX.

Through the experiences gained in fuel processing and mixed conversion, the PNC has been able to develop a plutonium handling technology, a fuel processing system, and a safeguard and weight control technology, and is putting that to use in preparing to build a new facility inside the No. 3 Plutonium Fuel Development FBR Line.

In order to achieve the goal of commercial plutonium use, the work will now shift toward achieving better cost and safety and increasing the amount of fuel that can be processed.

Technology for Plutonium Security Measures

94FE0521G Tokyo GENSHIRYOKU KOGYO in Japanese Jan 94 pp 60-64

[Article by M. Akiba, Power Reactor and Nuclear Fuel Development Corporation (PNC), Nuclear Material Control Division]

[Excerpts]

3. Security Measures at PNC Plutonium Handling Facilities

(1) Reprocessing Plants

The fuel reprocessing plants of the PNC are important facilities in the nuclear fuel cycle, which is about making effective use of nuclear material. These are the facilities that reprocess the spent fuel of light water reactors and advanced thermal reactors.

The nuclear material handling areas of these facilities are divided into three main categories, namely, spent fuel receiving and storage area, main processing area (chemical processing, analysis, and reagent conditioning areas), and storage area (plutonium product and uranium storage areas).

The quantity of plutonium in these facilities is verified by a receiving weigh-in tank and a shipping weigh-out tank. Thus, samples are taken from the weigh-in and weigh-out tanks at the reprocessing plants (facilities), and an analysis is done of the plutonium content to determine the quantity of plutonium in the tanks.

It used to take so long to get a result from a conditioned sample in the conventional isotopic dilution method that the PNC developed a new method which is more accurate and faster and enables measurements to be taken by mass spectrometry.

The weigh-out weighing is done with a K-edge densimeter, which is a non-destructive device for measuring plutonium content, and the weigh-in is done by isotopic dilution gamma spectrometry which also measures plutonium content.

(i) Isotope Dilution Gamma Spectrometry

Isotope dilution gamma spectrometry (IDGS) is a method that uses high resolution gamma spectrometry to measure the plutonium content and isotopic ratio in spent fuel dissolving solution (plutonium nitrate solution).

The plutonium in the sample is adsorbed and separated with an ion-exchanging resin, and the content of plutonium is found after mass having high ion-exchanging capability is reabsorbed and measured by gamma spectrometry.

The isotopic ratio of plutonium is found by taking mass containing more than 98 percent Pu and using it as a spike in the isotopic dilution. The dissolving solution is spiked with the sample and changes in isotopic ratio are measured by gamma spectrometry.

(2) Plutonium Fuel Plant

This plant is a large-scale facility having an MOX manufacturing capacity of 5 tons/year, and is outfitted with a host of remote-operated and robotic equipment for limiting worker exposure to radiation generated by the large quantity of nuclear material used. The plant also has a new safeguard system driven by advanced technologies tied to the automated equipment.

The technologies for this system were developed both independently and in joint research. After being installed at plutonium fuel handling facilities, it was used by both the Science and Technology Agency and the IAEA for inspections.

The nuclear material handling areas in this facility are divided into three different categories, namely, raw material storage areas, processing areas (pellet and fuel pin manufacturing), and assembly storage areas.

The safeguard system consists of an advanced weight control system that controls the flow of nuclear material throughout the entire facility, an advanced sealed-off

monitoring system for the raw material and assembly storage area, a remote-controlled non-destructive system for measurements in the raw material and assembly storage area and the processing area, and an inspection weight verification system that evaluates and analyzes weight and measurement control data of the facility.

(i) Advanced Weight Control System

This system consists of a central control computer, a processing control computer, and an equipment controller.

These computers are all connected on-line to form a system which collects most data on the movement of nuclear material from each processing area, and ascertains the amount in stock within the facility in real time for the purpose of criticality control and weight control and to present that data when necessary for inspection.

(ii) Advanced Sealed-In Monitoring System

1. Advanced Sealed-In Monitoring System for Raw Material Storage Area

This system is configured with sensors, a CCTV camera, a crane monitor, and gamma ray detector, and is used to videotape the movement of raw material canisters within the storage area.

2. Advanced Sealed-In Monitoring System for Assembly Storage Area

This is a system which employs a high-speed imaging technology to monitor the movement of fuel assemblies in real-time by color identification of the crane and assembly. This system not only prepares inventory maps of the storage area, but it also has recorded images of all the essential movements inside and outside the storage area.

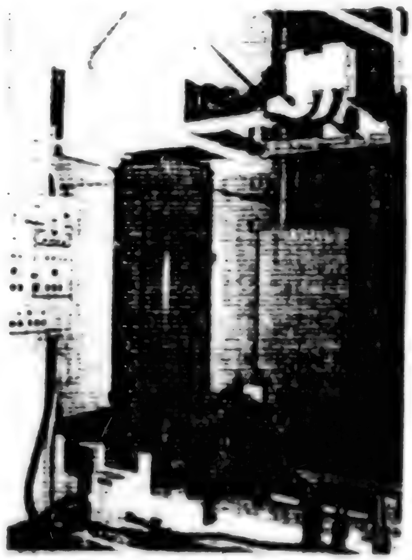
(iii) Remote-Control Non-Destructive Measurement System

This system was developed jointly by the PNC and DOE as part of the security cooperation agreement between the United States and Japan, and is used to verify plutonium content by measuring the amount of neutrons using the neutron coincidence method (a method of counting that distinguishes neutrons produced by simultaneous spontaneous fission of plutonium isotopes from the "background" released by neutrons and oxygen [α, n] reaction).

As we see below, the PCAS and FAAS have been installed in the storage area, and the FPAS, MAGB, and GBAS have been installed in the manufacturing process area.

1. PCAS (Plutonium Canister Assay System) (Photograph 1)

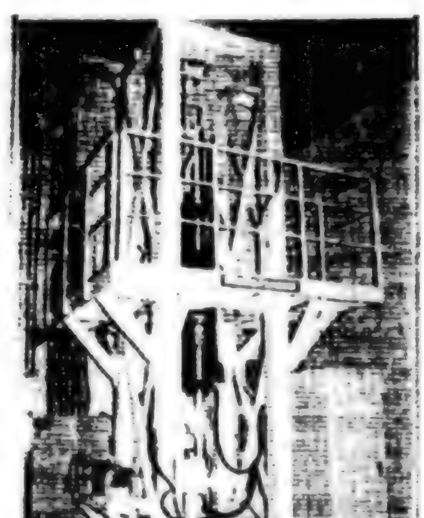
The canister counter (PCAS) was developed in order to quantitatively measure the plutonium content in the PuO_2 and MOX raw material storage containers (canisters). This system is built into the cylindrical automated canister conveyor located at the entrance/exit of the raw material storage area.



Photograph 1. Plutonium Canister Assay System (PCAS)

2. FAAS (Fuel Assembly/Capsule Assay System) (Photograph 2)

The capsule counter for assemblies (FAAS) is a system in which cylindrical detectors are installed at the exit/entrance of the assembly storage area. This system was developed in order to measure plutonium content quantitatively in the fuel assemblies shipped to the storage area.

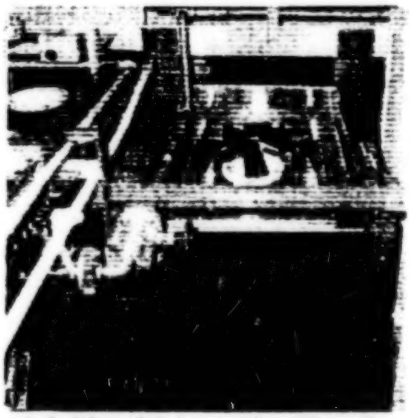


Photograph 2. Fuel Assembly/Capsule Assay System (FAAS)

3. MAGB (Material Accounting Glove Box) (Photograph 3)

The MAGB is a glove box used to measure plutonium content in transport containers (intermediate storage containers) by two slab-type detectors installed on the outside of the glove box that are used to measure the weight of nuclear material in the pellet manufacturing process. The transport containers contain either powder or pellets and are automatically transported from the intermediate storage areas to the glove box where they are weighed.

The MAGB-1 is used mainly to measure raw powder, the MAGB-2 to measure mixed powders and recovered fuel powders, and the MAGB-3 to measure sintered pellets.



Photograph 3. Material Accounting Glove Box (MAGB)

4. FPAS (Fuel Pin Assay System) (Photograph 4)

The FPAS is a system installed on top of the fuel pin tray conveyor to measure plutonium content in the fuel pins on the fuel tray.

This system is capable of measuring the plutonium content of the fuel pins of the "Monju" and "Joyo" FBR.

5. Glove Box Assay System (GBAS)

The hold-up counter (GBAS) is a system used to measure residual plutonium in the processing glove box. The system has two large slab-type detectors fitted into specially designed lifters that calculate plutonium weight by measuring the entire surface of the glove box.

The GBAS-1 is used mainly in the raw material processing, the GBAS-2 in the powder processing, and the GBAS-3 in the pellet inspecting process.

(iv) Advanced Weight Verification System

The nuclear material in the processing area is in bulk form, so a sealed monitoring system like the aforementioned advanced sealed-in monitoring system may not be used. Therefore, to maintain and monitor the continuity

Photograph 4. Fuel Pin Assay System (FPAS)

of nuclear material data, a near real time material accountancy system (NRTA) has been adopted.

(v) Other Safeguard Systems in Plutonium Fuel Plant

The Plutonium Fuel Plant has other safeguard systems that were developed through joint R&D. These include the INVS and HRGS which were installed as safeguard equipment for IAEA inspections.

The INVS (inventory sample assay system) is a system that finds plutonium content in a host sample by measuring the amount of neutrons in a smaller sample using the high-level neutron coincidence method.

The HRGS (high resolution gamma spectrometry system), on the other hand, is used to find plutonium content by measuring the gamma rays of plutonium isotopes in the smaller sample in order to verify the plutonium isotope ratio of the host sample.

(3) "Joyo," "Monju"

The experimental "Joyo" FBR uses Pu-U mixed oxide fuel, and the plutonium which it has in stock is only second to that of the Plutonium Fuel Plant and the Tokai Reprocessing Plant among PNC facilities. For that reason, the PNC has developed an effective method of verifying in-core fuel inventory and the spent fuel in storage at the spent fuel storage pool (being stored in drums), both of which need special measures taken in the area of security. There is really only one nuclear fuel material handling area in this facility.

The No. 1 Spent Fuel Pool, which is where spent fuel removed from reactors is initially put in storage, employs a "Churenkoff" optical monitoring system to

monitor spent fuel that must be cooled for approximately one year, but the No. 2 pool employs a radiation measuring method (GRAND-1).

The inspection criteria for evaluating initial in-core fuel states that it is necessary to verify up to one year's inventory, but because Pu-U mixed oxide fuel is used as fuel and sodium as coolant, verification by direct observation is not possible. A subsequent revision of the evaluation criteria, however, now states that inspecting fuel at the time of loading and removal will be acceptable.

That has led to a joint study between Japan and the IAEA on a method of verifying the movement of fuel, and another study on a device for measuring radioactivity.

This method of verification is called the Joyo in-core fuel verification system, and a similar such system is used by the prototype "Monju" FBR.

Below, we describe this system.

(i) In-Core Fuel Verification System

The fuel handling pathways are areas that concern inspectors from the view of security measures, but these are sealed-off areas which cannot be inspected directly, so an in-core fuel monitoring system employing 8mm video cameras has to be used to monitor those areas. This system consists of four verification monitors, namely, the ENGM which is used to monitor the new fuel storage area, the CCRM which monitors the cask car, the EVRM which monitors the fuel transfer machine, and the EXGM which monitors the No. 1 Spent Fuel Storage Pool.

1. ENGM (Entrance Gate Monitor)

The ENGM is a type of passive neutron coincidence collar that is able to discriminate and measure the number of spontaneous neutrons from Pu isotopes. This device is basically the same as the remote-controlled non-destructive measurement systems (PCAS, FAAS, etc.) used at the Plutonium Fuel Plant as verification equipment. This monitor is used in the "Joyo" to verify flow (movement) and gross defect of fuel.

The detection unit has a total of 24 built-in 3He detectors, six detectors to a side.

The ENGM is also to distinguish the direction in which the new fuel passes.

2. CCRM (Cask Car Radiation Monitor)

The CCRM uses one 3He detector for detecting neutrons, and one NAI detector for gamma ray detection. It also employs a gamma ray and neutron detector electronics package (GRAND-III) as equipment to verify fuel exchanges.

The CCRM can differentiate between fuel and other core components on the basis of whether neutrons are present or not, and can differentiate between new fuel and spent fuel based on neutron strength.

3. EVRM (Ex-Vessel Transfer Machine Radiation Monitor)

The EVRM is the same as the CCRM in terms of specifications and performance of the detectors and measuring equipment.

4. EXGM (Exit Gate Monitor)

With the EXGM, a sealed container housing a neutron and gamma ray detector is installed underwater using a wall rail and car of the spent fuel storage pool. The plant uses a 10B detector with weak sensitivity to gamma rays because of the effect that gamma rays have on neutron measurement due to the close proximity with which spent fuel, irradiated reflectors and control rods pass by the detectors.

Gamma rays are measured by an ion chamber, which is capable of detecting gamma rays at low radiation dosages in both spent fuel and new fuel. The gamma ray detectors are installed at the top and bottom, and are able to verify the direction in which spent fuel and other core components are pulled down from the cars.

4. Conclusion

The PNC, as we have just seen, operates facilities that handle large quantities of plutonium, and from that has gained valuable experience in terms of security measures. With the recent increased international concern over nuclear non-proliferation, these safeguard measures are being seen as a technology that will form a foundation to just such a system. Under such circumstances, the PNC believes it will be in a position in the future to build upon and strengthen safeguard technology while supporting the government and IAEA in maintaining and strengthening the international nuclear non-proliferation system.

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